Advanced Fuels Campaign 2018 Accomplishments

Heather D Medema, Kate M Richardson, Heather D Medema

November 2018

INL
Idaho National Laboratory

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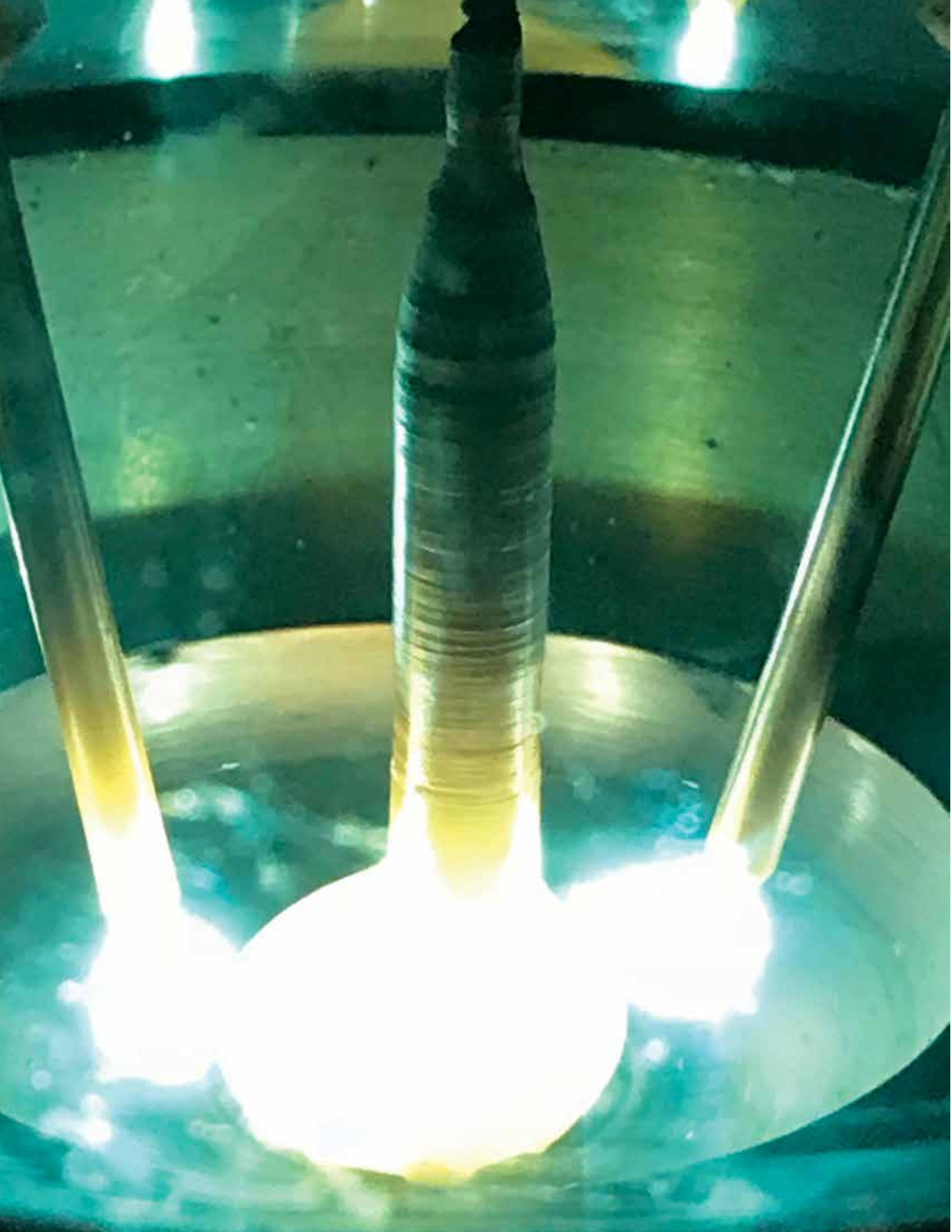
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ADVANCED FUELS CAMPAIGN
2018 Accomplishments

AFC Advanced Fuels Campaign

Fuel Cycle Research and Development

Advanced Fuels Campaign
2018 Accomplishments

INL/EXT
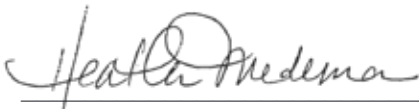
November 2018

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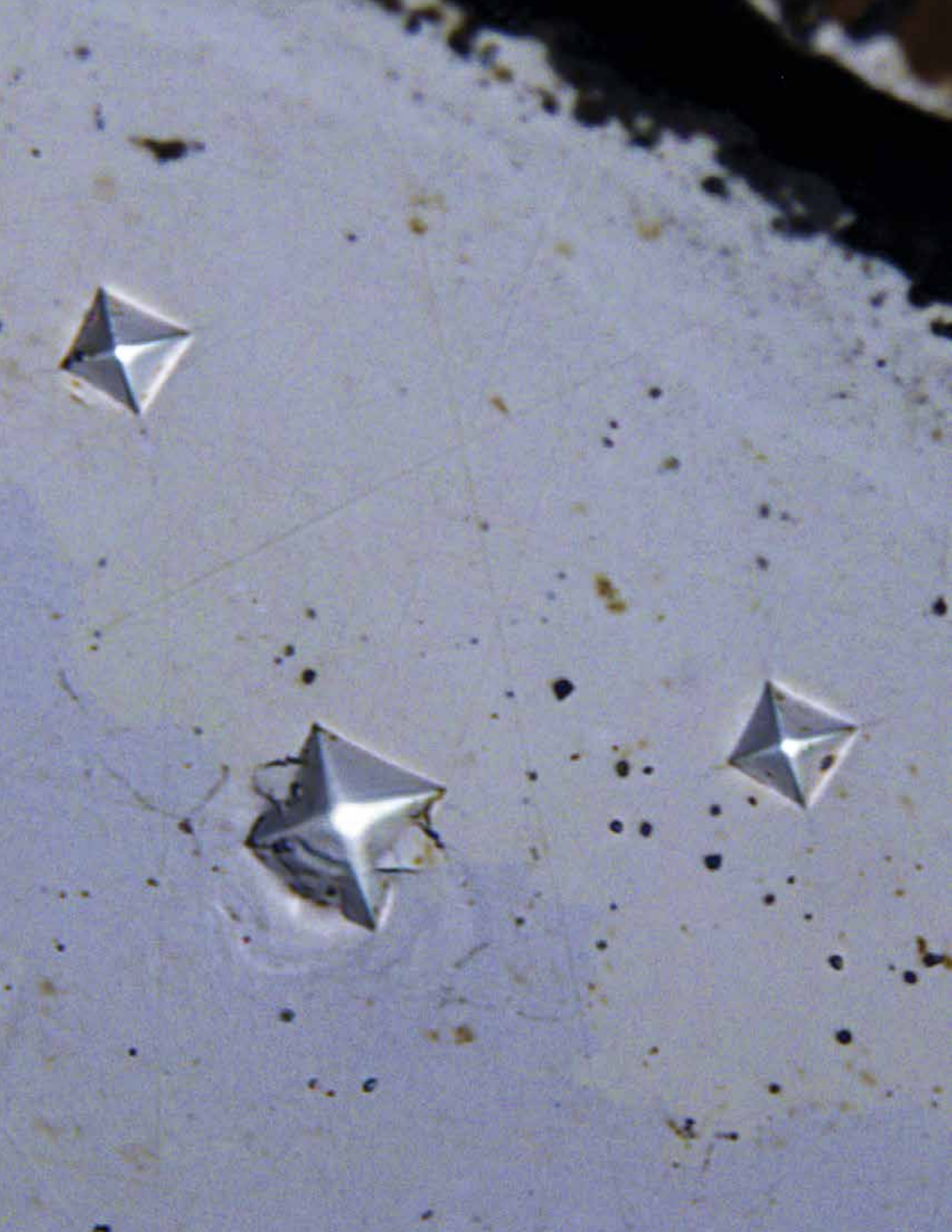
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ADVANCED REACTOR FUELS SYSTEMS

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1.2 FROM THE DIRECTOR

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The mission of the Advanced Fuels Campaign (AFC) is to perform research, development, and demonstration (RD&D) activities in support of identifying and developing innovative fuels and cladding materials and associated technologies to improve the performance and safety of current and future reactors; increase the efficient utilization of nuclear energy resources; contribute to enhancing proliferation resistance of the nuclear fuel cycle; and address challenges related to waste management and ultimate disposal issues.

AFC pursues its mission objectives using a goal-oriented, science-based approach which seeks to develop a fundamental understanding of fuel and cladding behaviors under conditions that arise during fabrication, normal steady-state irradiation, off-normal transient scenarios, and storage/disposal. This approach includes advancing the theoretical understanding of fuel behavior, conducting fundamental and integral experiments, and supporting the mechanistic, multi-scale modeling of nuclear fuels to inform and guide fuel development projects, advance the technological readiness of promising fuel candidates, and ultimately support fuel qualification and licensing initiatives. In the area of advanced tools for the modeling and simulation of nuclear fuels, the AFC works in close partnership with the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program, participating with NEAMS in developing mechanistic fuel behavior models and providing experimental data to inform and validate their most advanced tools.

Specifically, AFC objectives in the coming five year horizon include:

1. support the industry-led development of Accident Tolerant Fuel (ATF) technologies with improved reliability and performance under normal operations and enhanced tolerance during hypothetical accident scenarios, with implementation of lead test rods/assemblies of one or more ATF concepts in commercial reactor(s) by 2022;

2. lead research and development on innovative fuel and cladding technologies for applications to future advanced reactors, especially fast-spectrum reactors, including reactors that utilize both once-through and recycle scenarios;

3. continue the development and demonstration of the science-based approach, contributing to the establishment of a state-of-the-art research and development (R&D) infrastructure, necessary to accelerate the development of advanced fuel concepts; and

4. collaborate with NEAMS on the development and validation of multi-scale, multi-physics, and increasingly predictive fuel performance models and codes.

This report provides concise summaries of many of the significant AFC accomplishments made during FY-18. Of particular note are the following key accomplishments and their significance:

- An update to the original ATF Roadmap to Congress was provided to Department of Energy (DOE), which includes a vision for the deployment of the first batch reloads of one or more ATF concepts in commercial reactor(s) by 2025-2026.
- An alignment workshop was held between the AFC, Nuclear Regulatory Commission (NRC), and industry participants active in developing ATF concepts, the result of which was greatly improved communication and cooperation relative to activities needed to advanced ATF concepts toward eventual licensing activities.
- A gap assessment was completed, and mitigation strategies were developed, to address the shutdown of the Halden reactor and the important role it was expected to play in testing and qualification of ATF concepts.
- Early results from postirradiation examination (PIE) of Westinghouse and Framatome fuels from ATF-1 testing in the Advanced Test Reactor (ATR) were obtained, providing valuable insight into their performance as ATF concepts.
- The multi-year activity to establish a prototypic testing environment in the ATR was completed, and the ATF-2 pressurized water loop began operation with test fuels supplied by Westinghouse and Framatome and including standard oxide/zircaloy

reference pins to be conditioned for future Transient Reactor Test Facility (TREAT) experiments.

- A high burnup commercial fuel segment was refabricated in a hot cell environment and subjected to a simulated loss of coolant accident (LOCA) in the Severe Accident Test Station (SATS), demonstrating the successful reestablishment of a domestic LOCA testing capability that had been lost but will be needed for qualifying ATF fuel concepts.
- A new MiniFuel irradiation device was designed for the High Flux Isotope Reactor (HFIR), which will provide the capability for performing separate effects tests on ATF fuel concepts to inform and validate ATF fuel modeling activities.
- A new fabrication line capable of supplying silicide fuel pellets for the Westinghouse lead fuel rod project was established, and pellet production was in progress as the year came to an end.
- Characterization of the corrosion and hydriding performance of silicide fuel (U_3Si_2) was completed, which identified behaviors under hypothetical leaker fuel scenarios what will be important for developers to address.
- Significant advances were made in development and testing of oxide dispersion-strengthened (ODS) versions of FeCrAl alloys, including conventional and modified burst testing of these and other ATF cladding materials.
- The first transient experiment in the restarted TREAT reactor was performed, which included a fueled

ATF- Separate Effects Test Holder (SETH) capsule designed and fabricated by AFC.

- Fabrication of annular metallic fuels by extrusion was demonstrated, important advances were made in identifying and testing metallic fuel additives that could mitigate or eliminate fuel-cladding chemical interaction, and a new process was developed to rapidly produce small quantities of pure Am and Np metallic feedstock materials.
- Metallic fuels for the Integrated Recycle Test (IRT) were fabricated in a remote hot cell environment using actinide materials recovered from spent fuel, and irradiation of IRT-1 began in the ATR, which will soon provide important information on the performance of recycled metallic fuels.
- PIE results were obtained for annular, low smear density metallic fuels, showing very promising performance for future once-through, ultra-high burnup applications.
- A concept was developed for greatly increasing the burnup rate of nuclear fuels in the ATR, which along with advanced, mechanistic modeling offers genuine promise in accelerating the development and qualification of new fuels.
- The first release was made of the BISON fuel performance code with a mature and comprehensive capability to model metallic fuels, and accomplishment realized in cooperation with NEAMS.

1.3 SHOWCASE CAPABILITIES

The Transient Reactor Test (TREAT)

Principal Investigator: Dan Wachs
Collaborators: Todd Pavey

> **Experiment instrumentation and experiment handling, assembly and disassembly are vital to ensure an experiment is assembled, disassembled and handled correctly to ensure the instrumentation gathers the proper data from an experiment for comparison with other experiments.**

The Transient Reactor Test (TREAT) has proven that the base capabilities for performing experiments still exist. Improvements and new capabilities are required to enhance TREAT's capabilities for the future. The transient testing experiments program is actively pursuing the research development and demonstration of several advanced instrument technologies to meet near-term experimental programmatic goals while establishing the measurement capabilities for ongoing experimentation. Also, research and development continues on improving experiment assembly and disassembly capabilities at TREAT.

Project Description:
The objective of the TREAT instrumentation capabilities is to improve and qualify sensors for near-term, cross-cutting experimenter needs ranging from off-the-shelf instrument integration, experiment- and TREAT-specific needs such as unique sensors or sensor modifications, custom feedthroughs, instrument circuitry, etc. and provide a platform to integrate advanced sensors development at TREAT. One objective of the TREAT capabilities is to restore and

improve the capabilities at TREAT and Materials and Fuels Complex (MFC) to make experiment handling easier and safer for operations personnel. The physical test envelope in the TREAT reactor is unchanged and the basic handling equipment is unchanged from past operations. Experiments must remain compatible for transportation in the Hot Fuel Examination Facility (HFEF-15) cask. While future water loops may likely occupy a larger cross-sectional footprint than previous loops, it is assumed that handling of all assembled experiments in HFEF can be restored using the same handling fixtures or with minor modifications. Much of the capability to handle TREAT experiments has been lost and needs to be restored. Another objective of the TREAT capabilities is to restore and improve assembly and disassembly of experiments. Support has been required to restore capability to assemble static vessel test trains. The support included electrical, gas, and fluid interfaces for pressurizing vessels and testing the assembled static system to verify proper initial conditions and proper functioning of the test train instrumentation.

Accomplishments:

Instrument capabilities were improved by adding a new Experiment Instrument Panel (EIP) on top of the reactor that is connected to the Data Acquisition System (DAS) for use with experiments with instrument leads and heater controls completing a Level 2 milestone "Complete design and build experiment support equipment to support instrumentation and controls for each experiment." The EIP was designed and assembled by Eric D. Larsen. TREAT Maintenance personnel installed the new EIP and made the connections to the DAS plus completed System Operability (SO) testing to verify the EIP functions as expected.

Instrument capabilities were improved by installing instrumentation into the TREAT reactor during prescription testing. Installed into the core was micro-pocket fission detector (MPFD), self-powered neutron detectors (SPNDs) and optical fibers/pyrometers. Having the instruments in-core during operations has proven the instruments are very reliable. The instrument team included Colby Jensen, Austin Fleming, Kevin Tsai and the TREAT operations team.

Instrument capabilities were improved with a joint nuclear instruments testing in TREAT project with Massachusetts Institute of Technology (MIT). A sensor package consisting of four self-powered gamma detectors (SPGD), one SPND, one dual channel MPFD, one single channel MPFD, two type T thermocouples, and one type K thermocouple was installed into the TREAT reactor during temperature limited transients. The project was a proven success. The team included TREAT operations and experiment engineering personnel working with MIT students.

TREAT capabilities were proven by the assembly of five Separate Effects Tests Holder (SETH) capsules in Hot Fuel Examination Facility (HFEF) in preparation Accident Tolerant Fuel (ATF)-3-0 experiment campaign which commenced on September 18, 2018. The assembly included fresh fuel pins, welded thermocouples, fiber optic instrumented leads, filling capsule with helium gas, leak check, and radiography. The assembly team consisted of Nicolas Woolstenhulme, Devin Imholte, Doug Dempsey, Connor Woolum, Nathan Jerred, Lance Hone, Austin Fleming, and Aaron Craft.

TREAT capabilities were improved by designing and fabricating a shielded pig and stand for use to remove a SETH capsule from the primary can. The design was completed by Steve Swanson and Collin Knight; MFC fabrication shop fabricated the pig and stand. These proved to reduce the dose received to workers by a large amount allowing handling of the SETH capsule after the transient experiment.

Capability for Testing Light Water Reactor Fuels under Prototypical Conditions

Principal Investigator: Gary Povirk
Collaborators: Bryon Curnutt, Doug Crawford, Brian Durtschi, Kelly Ellis, Stephen Evans, Gary Hoggard, Paul Murray, Nate Oldham, Connor Woolum

Figure 1. A picture of the boron addition system, which was developed to control boron concentrations in the loop coolant to within specified levels.

The United States now has a dedicated facility available for the steady-state testing of commercial light water reactor fuels.

Following the disaster in Fukushima in 2011, the Accident Tolerant Fuels program (ATF) was initiated to develop improved fuel systems for light water reactors that would give reactor plant operators more time and flexibility when responding to an accident. Three fuel vendors, Framatome, General Electric, and Westinghouse responded to a call from the Department of Energy (DOE) with proposals for ATF concepts. Initial scoping tests were performed in capsule experiments in the Advanced Test Reactor (ATR; ATF-1), though it was always recognized that fuel qualification required a flowing loop with more prototypical coolant temperatures and water chemistries. Based on this need, loop testing in the central flux trap of the ATR (ATF-2) was conceived and executed. These steady-state tests are expected to provide important data for fuel qualification in their own right, but as importantly, ATF-2 specimens taken to specified burnups will be used for follow-on transient testing in Transient Reactor Test Facility (TREAT) and in the furnace at Oak Ridge National Laboratory (ORNL).

Project Description:

The ATF-2 loop test is considerably more difficult than the preceding ATF-1 capsule tests. The experiments required the development of systems to control water chemistry, holders and fuel pins that would allow for reconstitution of the test train during outages, and extensive discussions with the three fuel vendors to ensure that their programmatic needs were being met. In addition, insertion

of the experiment into the ATR necessitated nuclear, thermal hydraulic, and structural analyses and a test in the ATR critical facility to satisfy the requirements specified in the ATR Safety Analysis Report. Given that the experiment represents the first loop in the ATR to simulate commercial light water reactor conditions and the difficulties in meeting the competing priorities of the vendor teams, the experiment challenged the entire irradiation testing infrastructure at Idaho National Laboratory (INL), including the experiment design and analysis groups, the engineering staff at ATR, and the fabrication facilities at North Holmes Labs, Materials and Fuels Complex (MFC) and ATR. Their efforts have led to the development of the first experimental apparatus in the United States dedicated to the steady-state testing of commercial light water reactor fuels.

Accomplishments:

Success of the ATF-2 experiment required a substantial engineering effort. Methods to control coolant chemistry, including boron, lithium hydroxide, and dissolved oxygen and hydrogen concentrations were developed. As one example, Figure 1 shows the boron addition system that was designed and fabricated that saves weeks of operation time for each ATR cycle. The holders and test train (see Figure 2) were designed so that the fuel pins could be removed after
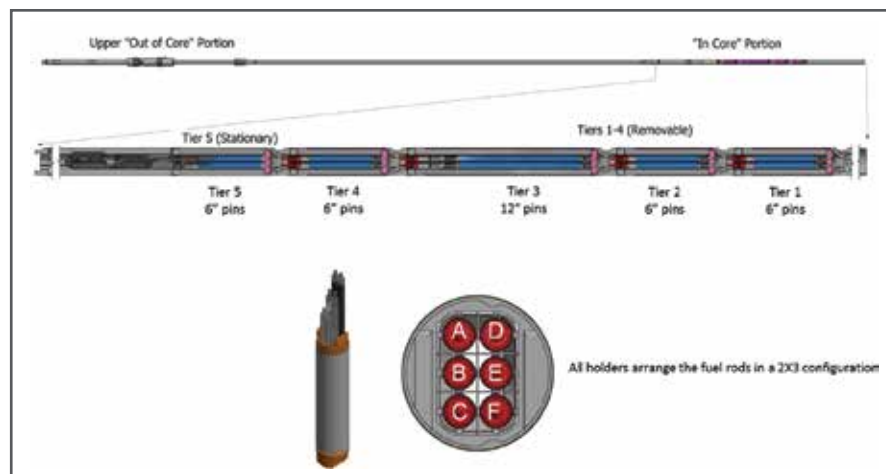
reaching a desired burnup and so the holder could be reconstituted with fresh fuel pins while sitting in the ATR canal during outages. The holders were fabricated either from stainless steel or from stainless steel with hafnium shrouds so that power levels were kept within specified bounds. Safety and programmatic analyses were performed to demonstrate that programmatic goals would be reached and ATR safety requirements were met. Difficulties in this regard included schedule pressures and design changes through the course of the project that caused significant amounts of re-analysis. Despite these challenges, the ATF-2 loop experiment was inserted into the ATR in June of 2018. The successful installation of the experiment is a testament to the dedicated teams at the fuel vendors and at the INL.

Figure 2. A schematic of the ATF-2 test train and fuel pin holders.

Development of an Irradiation Capability for 'Separate Effects' Testing of Nuclear Fuels: The MiniFuel Irradiation Design

Oak Ridge National Laboratory

Figure 1. Overview of the MiniFuel irradiation design. Each sub-capsule (shown at right) contains a single fuel sample (fuel kernels shown here). Six sub-capsules are loaded in each target (center right), and three targets can be loaded in each basket (center left). A total of three baskets can be placed in each VXF target position in HFIR.
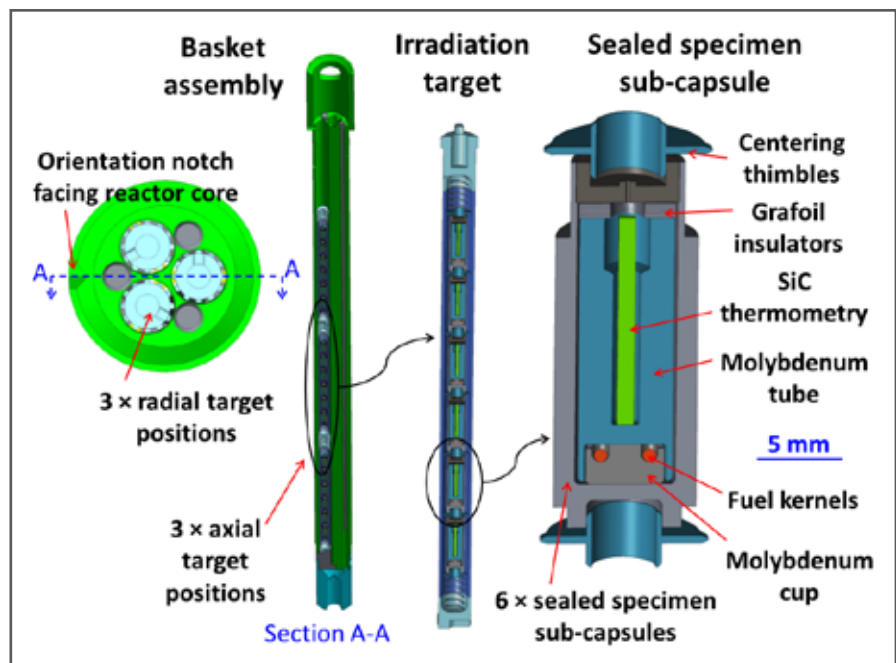
The challenge of qualifying new nuclear fuel materials is significant. A large dataset is necessary to ensure that reactor operators and regulators can predict fuel performance across all operating conditions and as a function of numerous variables. This data has traditionally been acquired through many integral fuel irradiation experiments. This process is not only expensive but also takes many years to travel from conception to design, fabrication, irradiation, and postirradiation examination. Recent years have also seen advancements in modeling and simulation capabilities. Our ability to model specific phenomenon across multiple length scales has greatly improved. However, the complexity of chemical and structural evolutions anticipated of nuclear fuels during operation poses a significant challenge. Integral irradiations convolute numerous interdependent effects; it has not been

possible for a researcher to isolate, for example, fission product mobility in an experiment without also assuming important details like temperature and microstructural evolution. The multifaceted drivers of cost, time, and complexity have prompted nuclear engineers and material scientists to explore alternatives to integral testing. The Advanced Fuels Campaign (AFC) has identified one potential solution using the High Flux Isotope Reactor (HFIR). The MiniFuel test is designed to overcome the traditional challenges of integral testing and offer researchers an efficient but powerful tool that supports a science-based approach to fuel qualification.

Project Description:

The MiniFuel design concept places fuel specimens inside individually sealed sub-capsules encapsulated by steel targets in the reflector of the reactor. Temperature is controlled by sizing an insulating gas gap between the sub-capsules and the target housing. Reducing the size of the fuel allows for very high fission rates (on a per unit mass basis) without prohibitively large temperature gradients. Furthermore, the small fuel mass results in the total heat generated in each sub-capsule being dominated by gamma heating instead of fission in the fuel itself. This essentially decouples the fuel temperature from the fission rate. This outcome is a critically important feature of the design to maximize utility for the modeling community; experiments can be performed where temperature gradients are less than 10 K/mm within the sample. Typical integral tests are over an order of magnitude larger.

The increased burnup rate that can be induced inside the fuel volume without resulting in extreme and non-representative temperature profiles and the thermal flux available in HFIR also provide another key advantage. Fuels can reach burnups typical to those of commercial light water reactors at discharge in roughly one calendar year (six cycles). Lower burnup capsules can be independently removed at shorter intervals. This means that researchers will be able to acquire the first postirradiation examination data far more rapidly than is possible for most integral tests.

Screening of new fuel concepts for disqualifying behaviors (e.g., significant swelling) can be accomplished in far shorter times. This may enable exploration of a broader range of fuel concepts and conditions than would otherwise be possible given the significant investment needed for test irradiations.

In addition to the benefits afforded by the irradiation conditions, samples do not need to be fabricated using enriched uranium. Depleted or natural uranium is adequate to achieve the burnup rates cited above. Despite the lower enrichment, very high powers can be achieved due to the extremely high neutron flux in the HFIR. Neutron absorption in 238U results in rapid breeding of 239Pu. Eventually, an equilibrium between 239Pu production and fission is achieved. At this point, the fuel fission rate remains nearly constant for the remainder of the irradiation.

Accomplishments:

The initial insertion of MiniFuel samples was achieved in FY18. This first irradiation is focused on baseline performance of particle fuels. The objective of this irradiation is to demonstrate the basic capsule build, gain familiarity with assembly and handling of fuel materials smaller than are standard, establish the basic ability of the MiniFuel capsule to achieve the desired irradiation conditions, and provide initial samples for development of postirradiation examination (PIE) techniques. Successful demonstration of PIE for small fuel samples is the major challenge of MiniFuel. While the small volume provides numerous advantages as detailed above, the reduced fuel volume does strain conventional methods for determination of critical fuel properties such as swelling and fission gas behavior. Modern tomography and automated software analysis have been developed to calculate sample volume

to the accuracy needed. Work in FY18 established the basic approaches for this method and established the theoretical error. Postirradiation examination of the first MiniFuel samples in FY19 will validate these analyses and provide insights regarding future improvements. Existing state of the art microscopy and small-scale test methods already in place at the Low Activation Materials Development and Analysis Laboratory (LAMDA) have already been used for characterization of irradiated fuel materials and will further augment PIE.

The ability of AFC researchers to fabricate test samples using depleted or natural uranium significantly brings down the cost of sample fabrication, increasing the number of variables that can be investigated. For example, uranium dioxide containing various dopants at differing levels can be irradiated to assess their impact on fission gas release. Traditional integral testing would require the each rodlet contain fuel pellets of a single chemistry. Constraints on irradiation capacity would limit AFC to only a few

different dopants every few years. The MiniFuel design provides the ability to study a wide range of dopants, as each sample is individually encapsulated. In a few calendar years, numerous dopants, concentrations, and irradiation temperatures can be studied.

Initial focus of MiniFuel irradiations in the coming years will be screening of fuel compositions and baseline data sets for the modeling and simulation communities. Interest from the broader nuclear fuels community in use of the MiniFuel capability for screening studies has also been significant. Once fully demonstrated and optimized, the MiniFuel irradiation will position AFC to perform unprecedented investigation of nuclear fuel behavior at a cost and throughput rate not possible using traditional testing.

Uranium Silicide Fuel Pellet Production

Idaho National Laboratory

Figure 1. The new fuel fabrication line, located in the Experimental Fuels Facility at Idaho National Laboratory features an arc-melter, sintering furnace, and an inert high-density fuels glovebox.

As an alternative to the conventional uranium dioxide (UO2) fuel system, a team at Idaho National Laboratory (INL) has developed a new capability to manufacture uranium silicide (U3Si2) fuel pellets for the Westinghouse Electric Company as part of their EnCoreTM product line of accident tolerant fuels.

Manufactured in the Experimental Fuels Facility (EFF) at the Materials and Fuels Complex (MFC), the advantages of the U3Si2 fuel is a higher uranium density and much higher thermal conductivity. With a higher thermal conductivity, the EnCore fuel promises a safety improvement by tolerating a much higher linear heat rate prior to fuel melt. The EnCore

Figure 2. The new fuel fabrication line, located in the Experimental Fuels Facility at Idaho National Laboratory features an arc-melter, sintering furnace, and an inert high-density fuels glovebox.

fuel system, when fully deployed, has been characterized as a game changer for the nuclear industry due to significantly increased safety margins in severe accident scenarios. The economic benefits are estimated to be on the order of hundreds of millions of dollars.

Producing a new fuel for Westinghouse and its customer Exelon Nuclear has allowed INL to contribute directly to improving the safety and economics of commercial nuclear power generation. Exelon operates the largest fleet of nuclear plants in the country with 23 reactors located at 14 sites. The fuel pellets fabricated at

INL will be inserted into the Byron-2 commercial nuclear reactor in the spring of 2019, completing a very important milestone on the path to eventual licensing of the entirely new fuel system. This success with private nuclear industry is an important milestone for INL and the mission of the Advanced Fuels Campaign.

Experimental Fuels Facility

The EFF houses a diverse array of fuel fabrication capabilities supporting customers in the U.S. Department of Energy's (DOE's) Office of Nuclear

Figure 3. The new fuel fabrication line, located in the Experimental Fuels Facility at Idaho National Laboratory features an arc-melter, sintering furnace, and an inert high-density fuels glovebox.

Energy (NE) and private industry partners through INL's cooperative research and development program.

In order to fabricate the U3Si2 fuel pellets for the Westinghouse EnCore fuel pins, an entirely new fabrication line was built, tested, and commissioned at the EFF in 2018. This line consists of a large inert atmosphere glovebox, arc-melter, and sintering furnace (See Figure 1), as well as other powder processing support equipment. Aside from the Westinghouse/Exelon project, the fabrication line will be used to support the development and fabrication of other high density fuel concepts for years

to come. The arc-melter has already been used in the TerraPower project in preparation of some of its fast reactor metallic fuels for irradiation testing.

With the goal of sustaining nuclear power as a clean energy source in the U.S. while enhancing safety for the public, the U.S. Congress has renewed funding for the development of new nuclear fuels with enhanced accident tolerance and increased performance. The Nuclear Regulatory Commission (NRC) and DOE are collaborating to enhance the safety of reactors worldwide via the Accident Tolerant Fuel (ATF) Program.

ADVANCED LWR FUELS

2.1 ACCIDENT TOLERANT FUELS

ATF Industry Advisory Committee

Committee Chair: Bill Gassmann, Exelon
Collaborators: Steven Hayes, Ed Mai, Kate Richardson

The Advanced LWR Industry Advisory Committee (IAC) was established in 2012 to advise AFC's National Technical Director on the direction, development, and execution of the campaign's activities related to accident tolerant fuels for commercial light water reactors. The IAC is comprised of recognized leaders from the commercial light water reactor industry. They represent the major suppliers of nuclear steam supply systems, owners/operators of U.S. nuclear power plants, fuel vendors, EPRI, and NEI. Members are selected on the basis of their technical knowledge of nuclear plant and fuel performance issues as well as their decision-making authority in their respective companies. During the past year the committee provided important industry perspectives relative to the campaign's interactions with the Nuclear Regulatory Commission on issues related to activities supporting the eventual licensing of accident tolerant fuels and the ramifications of the Halden Reactor shutdown.

The IAC meets monthly via teleconference and is currently chaired by William Gassmann of Exelon Corporation. Additional members represent Westinghouse Electric Company, Global Nuclear Fuels, AREVA, Dominion, Duke Energy, Southern Nuclear, EPRI, and NEI.
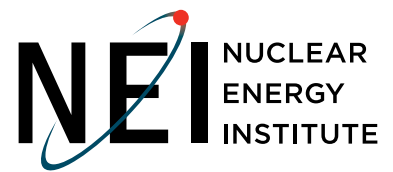
Accident Tolerant Fuel (ATF) Industry Teams – Westinghouse Electric Company LLC

Principal Investigator: E. J. Lahoda

Collaborators: Westinghouse Electric Company LLC, General Atomics (GA), Massachusetts Institute of Technology (MIT), Idaho National Laboratory (INL), Los Alamos National Laboratory (LANL), Exelon Nuclear, University of Wisconsin (UW), National Nuclear Laboratory (United Kingdom) (NNL), Army Research Laboratory (ARL)/VRC/MOOG, University of Virginia, University of South Carolina, Oak Ridge National Laboratory (ORNL), Fauske & Associates, Rensselaer Polytechnic Institute (RPI), University of Texas at San Antonio

Figure 1. Improved U_3Si_2 Pellets Manufactured by INL

EnCore fuel will provide not only significant fuel cost savings, but also enable operating cost savings for current nuclear plants due to the reclassification of safety related equipment and testing requirements to lower safety classifications due to enhancement of the ability of the fuel to better withstand beyond design basis events.

The overall objective of this program is to introduce EnCore®*, Westinghouse's accident tolerant fuel (ATF), lead test rods and lead test assemblies (LTRs/LTAs) for coated zirconium cladding with ADOPTTM* (a doped UO2) and UO2 and SiGATM** SiC with U_3Si_2 fuel into commercial reactors by 2019 and 2022, respectively. The objective of the current work is to design, test and build LTRs using commercially scalable technologies for up to 6 year-long exposure at pressurized water reactor (PWR) conditions using U_3Si_2 and Cr coated Zr alloy cladding. Test programs at MIT and ATR support the LTR work.

The data from this 6-year test reactor exposure and test evaluation will be used as the basis to license and load LTAs into commercial reactors.

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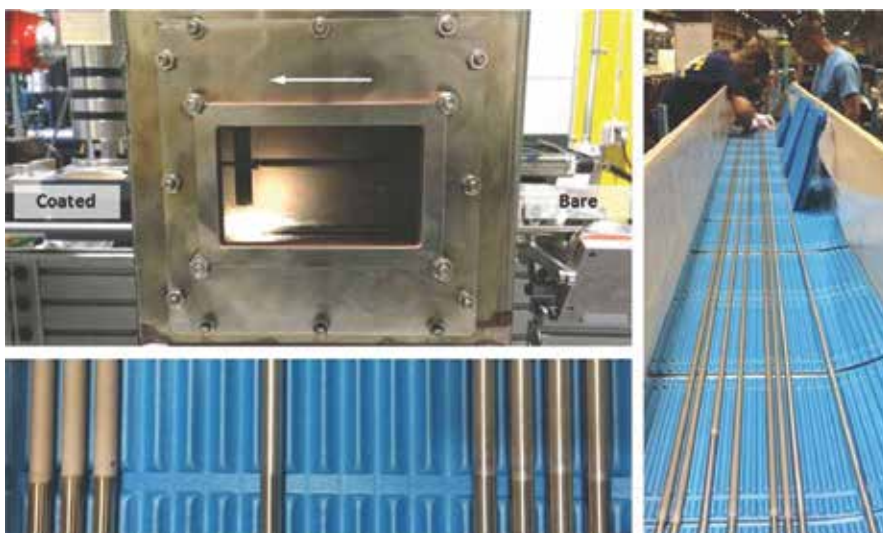
Figure 2. Manufacturing Cr Coated Zr Alloy Rods for Byron-2 Lead Test Rod

Project Description:

The technical objectives of this program are:

- Fabrication of representative test fuel rodlets consisting of SiC and Cr coated zirconium cladding with U_3Si_2 and ADOPT fuel and lead test rods consisting of Cr coated zirconium cladding with ADOPT and UO_2 fuel for testing and data generation

- Development of an ATF LTR/LTA project plan

- Generation of test data and modeling supporting test and commercial reactor feasibility

- Determine the expected behavior of the ATF options in loss of coolant accident (LOCA)/station blackout scenarios

- Evaluate the design basis and beyond design basis accident performance, expected plant licensing impacts and resultant economic savings for commercial plants

- Determine the feasibility of licensing a commercially feasible ATF

Meeting these objectives will move the nuclear industry closer to a nuclear fuel with enhanced tolerance

Figure 3. Improved SiC cladding by General Atomics – 1 Meter Tubes, Improved Composite Structure and Hermetic Endplugs.

of beyond design basis events and significantly lower fuel cycle costs for the utilities and their customers.

Accomplishments:

Rapid progress has been made on developing and producing the components for the LTRs to be inserted in Byron-2 in spring of 2019. These LTRs consists of segments containing U_3Si_2 pellets in thick-walled Zr tubes and Cr coated Zr alloy tubes containing both ADOPT and UO_2 pellets. The U3Si2 pellets were made by INL (Figure 1) and are much improved (few cracks and surface fazing) over previous pellets due to the use of new tooling. The ADOPT fuel will be made by Westinghouse Sweden and the coated cladding by ARL/VRC/MOOG (Figure 2). These components will be assembled at Westinghouse Columbia in the fall of 2018.

Progress continues on developing the components for the LTA to be introduced in 2022. This LTA will consist of Cr coated Zr alloy and SiC cladding. The fuel pellets will be oxidation resistant U_3Si_2, UO_2 and ADOPT. Significant progress has been made in the fabrication and characterization of engineered, multi-layered SiGAtm SiC-SiC-based accident tolerant fuel cladding. Improvements in dimensional control techniques allow for cladding fabrication meeting tolerances of +/-0.001 inch for a given target dimension, while controlling surface roughness to Ra of ~0.5 μm for the outer tube surface and Ra of ~10 μm on the inner surface. Rodlet assembly trials were performed using SiC-SiC composite tubing, springs, pellets, spacers, and thermometry components, and several improvements have been implemented to enable more reliable sealing of SiC-cladding rodlets with an internal helium backfill. Rodlet leak rates better than 1E-9 atm-cc/sec have been measured, which is well below the leak rate specification. New high-temperature characteriza-

tion methods were demonstrated, and SiGA-TM SiC-SiC cladding retains room temperature strength through 1650°C, and >65% strength to 1900°C, performing significantly better than other ATF concepts, that can show substantial strength loss at 600-900°C. Cladding thermal conductivity was measured up to 1300 °C, with values up to ~30 W/m-K at room temperature. In collaboration with the University of South Carolina, methods were developed to locally heat specimens beyond 1000°C to simulate pellet cladding mechanical interactions and/or perform quench testing. High temperature testing to simulate accident conditions are being pursued in the Westinghouse Ultra High Temperature facility (Figure 4).

Approaches are being developed at a variety of universities and national laboratories and tested for increasing the oxidation resistance of U_3Si_2. The various options will be tested in the leaker rod test apparatus developed by Westinghouse to verify effectiveness.

The commercial introduction of coated cladding with U_3Si_2 is targeted for 2023 and for SiC with U_3Si_2 for 2027. In order to meet these very aggressive target dates, changes in the way that Westinghouse and the utilities will license ATF with the Nuclear Regulatory Commission (NRC) will be required. Interactions with the NRC have been initiated and a modified licensing approach utilizing advanced modeling and fuel performance monitoring technologies is being pursued.
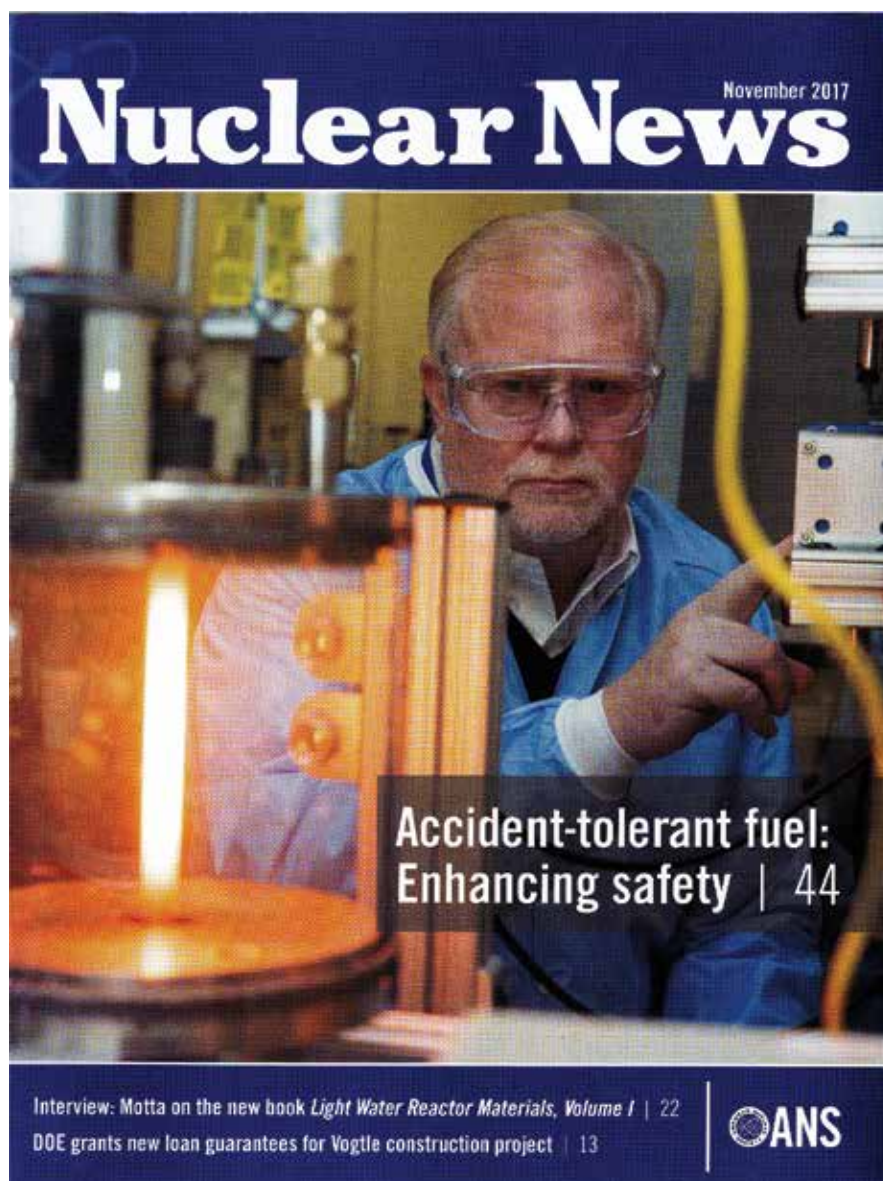

Figure 4. Westinghouse Ultrahigh Temperature (>2000°C) Steam Test Apparatus for ATF Cladding

ATF Industry Teams - Framatome

Principal Investigator: Kiran Nimishakavi

Cr-coated M5 sample:

54097-528-20-S.F 1100° - 18000s

Uncoated M5 reference sample:

54097-486-20-S.F. 1100° - 12000s

Figure 1. Photograph of cladding samples after 1100°C steam oxidation and quench: a) Cr-coated sample exposed for 18000s, b) uncoated sample exposed for 12000s.

The Framatome Team is investing significant research and development efforts to develop technologies which increase nuclear safety by improving fuel and containment integrity during design basis or beyond design basis events. As part of this effect, Framatome is developing near-term solutions that strengthen Framatome's existing product portfolio and a long-term, revolutionary new product which has the potential to support significant gains in safety and operating margins.

Project Description:

The ultimate goal of Department of Energy (DOE's) Enhanced Accident Tolerant Fuel (EATF) program is to develop an improved and more robust nuclear fuel design that will reduce or mitigate the consequences of reactor accidents and improve the economics of the reactors thermal output and reliability. After extensive testing, evaluation and downselection during the program's first phase, Framatome's Phase II technical approach addresses three focus areas: (i) Chromium (Cr)-coated cladding, (ii) Chromia-doped UO_2 fuel pellets, and (iii) silicon carbide (SiC) composite cladding.

A dense Cr-coating on a zirconium-based cladding substrate has the potential for improved high temperature steam oxidation resistance (see Figure 1) and high temperature creep performance, as well as improved wear properties. Over the course of the EATF program, extensive processing and testing activities are being carried out with the objective of fulfilling Lead Test Assemblies (LTAs) by 2019 and batch implementation by the mid-2020s.

Figure 2. Full length Cr-coated zirconium-based cladding.

At the fuel level, chromia-doped UO_2 pellets can improve the corrosion and fragmentation resistance over today's UO_2 fuel. To date, the performance of this fuel has been extensively studied in out-of-pile and in-pile test programs and modifications are being implemented to accommodate chromia-doped fuel in Framatome's fuel performance code.

For revolutionary (over-the-horizon) performance improvements, Framatome is developing a composite cladding comprising a silicon carbide fiber in a silicon carbide matrix (SiCf/ SiCm). The objective is to develop a system which does not suffer from the same rapid oxidation kinetics of zirconium-based cladding while having attractive operating features

such as reduced neutron absorption cross-section and higher mechanical strength at accident temperatures.

Accomplishments:

The Framatome Team has made significant technical progress towards these objectives in Government Fiscal Year 18.

For the Cr-coated cladding, out-of-pile testing has shown significantly reduced oxide growth when compared to uncoated cladding under identical LWR coolant conditions, demonstrating improved corrosion performance and post-quench ductility. Accelerated wear testing demonstrated superior tribological properties for the Cr-coating over uncoated Zr-based cladding, with

Figure 3. Cr coated cladding rodlets for ATR – ATF-2 irradiation.

For chromia-doped UO_2 fuel, pellet qualification testing has been completed on a full-scale production line at Framatome's Richland, WA manufacturing facility and upgrades are being made to additional UO_2 production lines for expansion.

The Framatome Team is preparing for Cr-coated cladding and chromia-doped pellet LTAs to be inserted in Vogtle-2 in Spring 2019 (as well as Cr-coated Lead Test Rods (LTRs) to be inserted into a Babcock & Wilcox (B&W) reactor by Fall 2019) and continues to perform manufacturing and design activities for this purpose. For the Cr-coated cladding, a coating deposition system has been built, tested, optimized, and qualified for full-length tubes. Coating uniformity (visual and thru-thickness) has been achieved over the full length of the tubes which are shown in Figure 2. To instill confidence in the full-length fabrication scale up efforts, high temperature air oxidation tests on tube segments showed behavior identical to samples coated on a smaller, prototype deposition system.

To support LTR/LTA insertion, simulation capabilities for chromia-doped fuel pellets have been added to Framatome's fuel performance code. A plan to support the batch implementation of the Framatome's chromia-doped fuel and Cr-coated cladding has been developed and presented to the Nuclear Regulatory Commission (NRC).

significant reduction in cladding wear volume. These measured properties show potential for improved in-service corrosion and fretting performance of cladding under normal and Loss of Coolant Accident (LOCA) conditions.

Figure 4. Samples of SiCf/SiC composite tubes (filament winding) after external / internal smoothing (© CEA).

Framatome's approach is to maximize levels of safety by addressing both the fuel and the cladding together through multiple, synergistic technologies for near-term evolutionary and longer-term revolutionary solution.

In June 2018, Framatome's EATF rodlets were inserted into Idaho National Laboratory (INL's) Advanced Test Reactor (ATR), representing a major milestone and the first opportunity to evaluate the combined performance of Cr-coated cladding with chromia-doped UO_2 pellets. The rodlets for this test program were fabricated in Framatome's Richland facility and are shown in Figure 3. Successful completion of this irradiation and testing program will demonstrate concept performance.

Significant progress has also been made towards the development of a SiCf/SiCm cladding shown in Figure 4. A suitable manufacturing process and fiber architecture has been identified and a mechanical tensile-test facility adapted to tubular silicon carbide composites has been developed. Preliminary mechanical properties are encouraging. A new recession-erosion test loop has been developed to test various solutions for mitigating SiC recession (corrosion) in representative LWR conditions. Tests are planned to address other key technical issues related to hermeticity, joining and manufacturing.

General Electric Development of Accident Tolerant Fuels

Principal Investigators: Raul B. Rebak and Russ M. Fawcett
Collaborators: Evan Dolley, Chuck Paone, Rich Augi, Myles Connor, Dan Lutz, Patrick Davis, Sarah Desilva, Marty Swan, Yang-Pi Lin

GNF's IronClad & ARMOR Accident Tolerant Fuel Development Team
First to Deploy Lead Test Assemblies in Commercial Reactor, Plant Hatch 2018

Figure 1. The GE Nuclear and Southern Nuclear Team which won the 2018 TIP award from the Nuclear Energy Institute (NEI) for the first commercial nuclear power plant to load accident tolerant IronClad and ARMOR fuel.

The objective of the current Phase 2 (Development and Qualification) General Electric (GE) project is to advance the design and development of nuclear fuels for light water reactors with enhanced accident tolerance. The project aims at demonstrating the feasibility of using FeCrAl alloys or IronClad to make commercial power reactors safer to operate. Besides IronClad, the project also includes a coating program for zirconium alloys called ARMOR and the study of cermet fuels.

Project Description:

The GE (including GE Global Research, Global Nuclear Fuels and GE Hitachi Nuclear) project involves the participation of three National Laboratories: Oak Ridge National Laboratory (ORNL), Idaho National Laboratory (INL), and Los Alamos National Laboratory (LANL) as well as the reactor owner utilities Exelon Generation and Southern Nuclear. The entire organization is working to meet the initial metric requirements from the Department of Energy (DOE) to make the reactors safer and more

economic to operate. The approach to the project is to enlist and integrate findings from the basic experimental and simulation research tools of the national laboratories to the fuel design and fabrication capabilities of Global Nuclear Fuels to the practical engineering skills of the commercial electrical power companies.

Accomplishments:

The Phase 2a (2017-2018) was very successful and included the insertion of IronClad and ARMOR components in the Hatch 1 nuclear plant in February 2018. The main goals of Phase 2b (2019-2021) are (1) to carry a second insertion into a commercial reactor, which is the Exelon Generation Clinton Plant in the Fall of 2019 and (2) To evaluate the initial post neutron irradiation examination of IronClad and ARMOR rods from the INL Advanced Test Reactor (ATR) and from the Hatch commercial power plant. Significant efforts were invested in the tube making process of FeCrAl alloys both at Sandvik and at US plants such as Superior Tube and Century Tube. Yukinori Yamamoto from ORNL was the lead in the success of the tube effort.

Two primary alloy producers (Metaltek in Wisconsin and Sophisticated Alloys in Pennsylvania) are fabricating ingots of C26M for tube production purposes.

Steam testing showed that C26M alloy is as resistant to attack by steam as APMT. High temperature ($\sim$300°C) hydrogenated and oxygenated water electrochemical testing showed that C26M and APMT behave in a similar manner as well-known reactor materials such as nickel alloy X-750 and stainless steel type 304.

GE/Global Nuclear Fuel (GNF) is working with utilities such as Southern Nuclear to insert the first FeCrAl fuel prototype into a commercial reactor in the US. The plant selected for the first effort is SNC's Plant Hatch, Unit 1 during the Cycle 29 refueling outage (1Q18). It is planned to insert 2-8 Lead Fuel Rods (LFRs) into each of 2-4 GNF2 Lead Test Assemblies (LTAs) as segmented rods of the two FeCrAl alloys mentioned before (APMT and C26M).

GNF/GE-Hitachi (GEH) and Southern interfaced with the Nuclear Regulatory Commission (NRC) to obtain the authorizations to transport the fuel to the Georgia plant and to install the LTA into the reactors.

The GE ARMOR and IronClad concepts for Accident Tolerant Fuel (ATF) are the most near term retrofit solutions for Generation III reactors involving minimal changes in reactor configuration.

Experimental Investigation and Simulation of Coated Cladding and Doped Fuel

Collaborators: K. Shirvan (Massachusetts Inst. of Tech.), K. Sridharan (U of Wisconsin), L. Shao (Texas A&M University), M. Tonks (U of Florida) J. REED (FRAMATOME), W. Liu (Structural Integrity), J. Hales (INL)

Figure 1. From left to right: Cold spary cr coated rodlet (including endcap) before testing; suspended in quartz tube at 1200oC; after 30 min; after 90 min where a defect was present in the coating showing extensive crack propagation; after 90 min where no defect was present in the coating showing rod bowing.
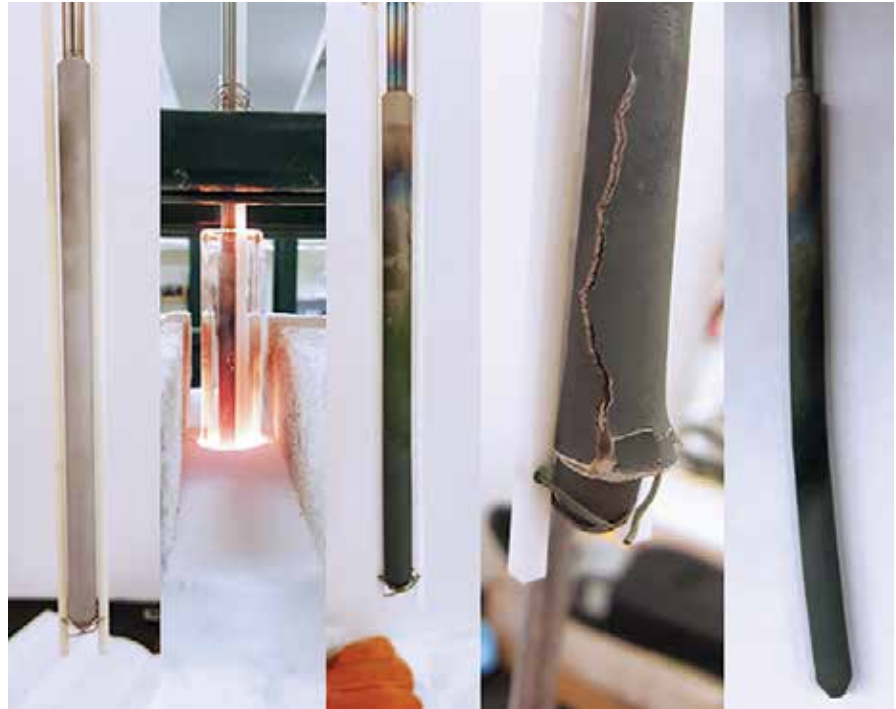
This integrated research project (IRP) focuses on experimental and modeling of time to failure for near term accident tolerant fuel (ATF) concepts. The IRP builds upon strong university capabilities at the Massachusetts Institute of Technology (MIT) with its experience in fuel design, testing and safety analysis, the University of Wisconsin (UW) with its experience in severe accident modeling and development of cladding coatings for the ATF industrial campaign, Texas A&M University (TAMU) with its material ion irradiation capability, and University of Florida (UF) with its meso-scale fuel performance modeling experience. Idaho National Laboratory (INL) is a member of the team as well, to allow for efficient implementation and integration of the models. The IRP also benefits from close collaboration with two industrial partners: Structural Integrity Engineering Firm with state-of-art experience in fuel modeling under accident conditions and FRAMATOME, one of the main nuclear fuel suppliers in the U.S.
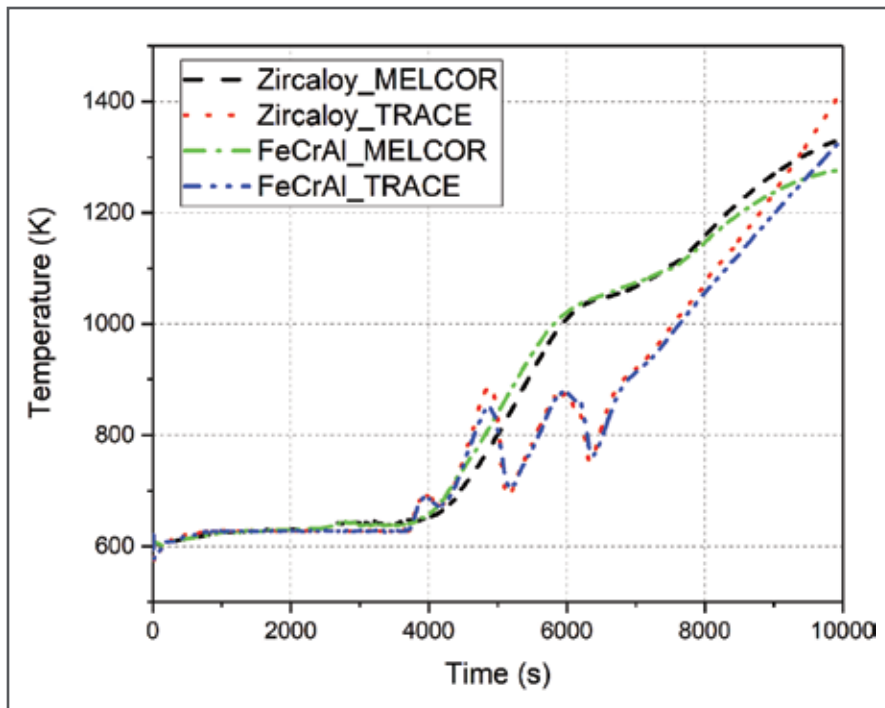
Figure 2. Comparison of MELCOR and TRACE prediction of peak cladding temperature during a station blackout scenario for a PWR benchmark problem with Zirc-4 and FeCrAl claddings.

2018 Highlights

The experimental work at MIT focused on testing chromium (Cr) coated samples supplied by outside contractors to investigate its performance. While the coating showed its ability to protect against diffusion of oxygen in steam environment, new failure modes were characterized through our test campaign specific to cold spray Cr. These included rod bowing at prolonged exposures to high temperature steam due to formation of a very thin oxide layer and accelerated crack propagation at presence of stress compared to Zr-4. Furthermore, quench heat transfer of Zr-4, Cr, FeCrAl and Mo were measured under as fabricated, oxidized and gamma irradiated conditions. The underlying mechanism behind impact of gamma irradiation in enhancement of quench heat transfer was also studied. TAMU has started ion irradiation of MIT Cr-coated samples to study Cr structural stability and mechanical performance of coating after and before irradiation. UW-Madison completed sample

Figure 3. Microstructure created by design tool using mesoscale simulation and macroscale modeling that will determine the most efficient use of a given additive fraction to maximize the increase in thermal conductivity.
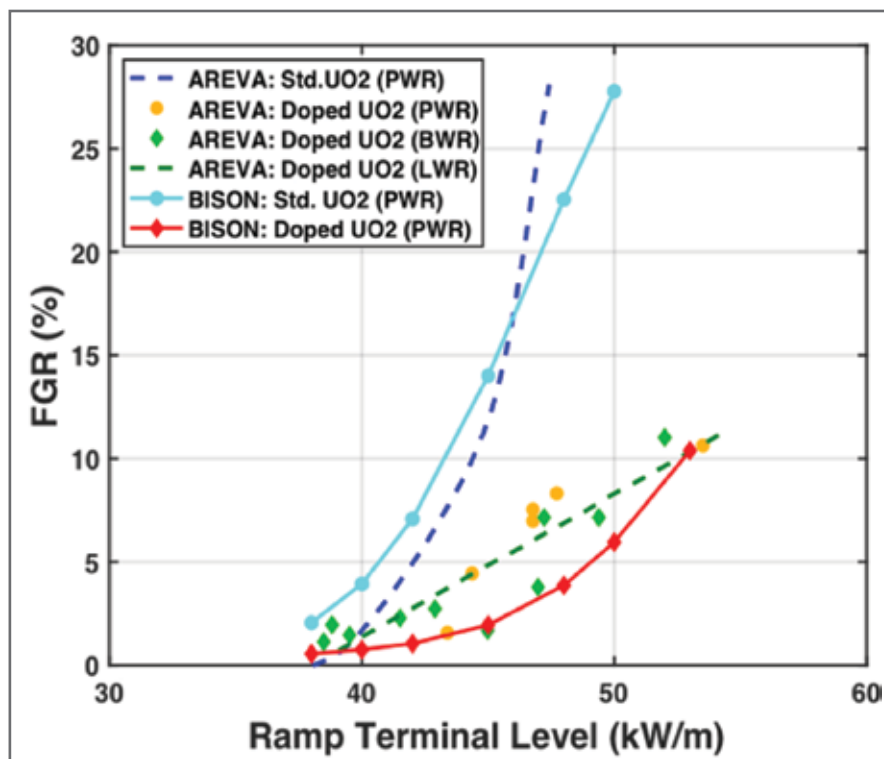
production this year. This process included substrate preparation, application of coatings, finishing of coatings, and measurements and documentation of each sample before shipment to MIT for testing. Following testing, MIT has been returning samples to UW-Madison for characterization. Characterization includes topological and cross-sectional Scanning Electron Microscopy (SEM), energy dispersive spectroscopy (EDS; compositional analysis), x-ray diffraction (XRD), and other techniques as needed.

On simulation side, the system code modeling with MELCOR and TRACE has continued. MIT and UW successfully finished a TRACE/MELCOR short

term station blackout benchmark to gain confidence in each system code prediction of the increased gains in additional coping time provided by the ATF concepts. In addition to pressurized water reactors (PWRs), MIT and UW have extended both code accident analysis to boiling water reactors (BWRs). The new results thus far still show notable but modest gains in coping time. The fuel performance of Cr-coated Zircaloy and Cr-coated SiC/SiC composite claddings during normal operation, power ramp and loss of coolant accident were performed with INL's BISON tool by MIT. The rod ejection simulations of the coating with BISON was performed by Structural
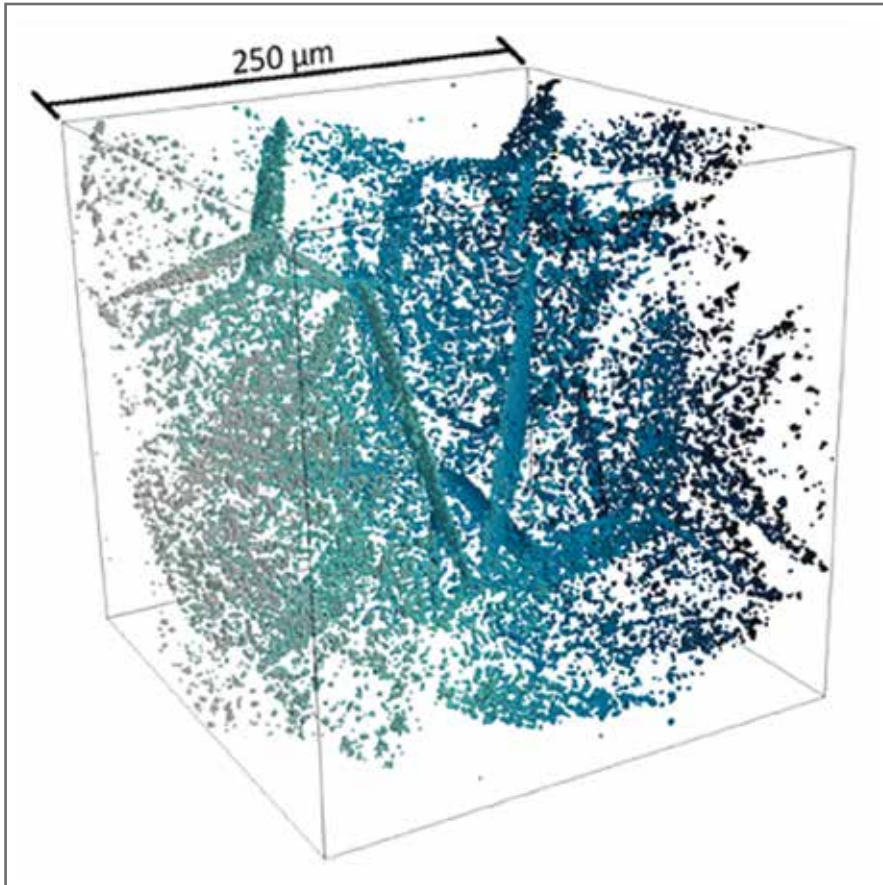
Figure 4. BISON prediction of fission gas release (FGR) for power ramps for conventional UO2 and Chromia doped UO2 vs. Experimental data.

Integrity. The results showed Cr is a promising concept, however, plastic deformations were predicted during normal operation that requires future experimental investigation. Another near-term ATF concept we are investigating is the addition of high thermal conductivity additives to UO2 reactor fuel. UF is creating a design tool using mesoscale simulation and macroscale modeling that will determine the most efficient use of a given additive fraction to maximize the increase in thermal conductivity. On

Chromia Doped fuel performance, MIT in collaboration with FRAMATOME and INL has benchmarked BISON performance against experimental data from Halden and FRAMATOME ramp tests. Detailed sensitivity study of doped fuel during power ramp and loss of coolant accident shows improved fission gas retention, softer pellet-to-clad interaction, delayed rod burst and smaller rod burst size.

OECD-NEA Expert Group on Accident Tolerant Fuels for LWRs

Principal Investigators: Kemal Pasamehmetoglu
Collaborators: Shannon Bragg-Sitton

The Organization for Economic Cooperation and Development /Nuclear Energy Agency (OECD/ NEA) Nuclear Science Committee approved the formation of an Expert Group on Accident Tolerant Fuel (ATF) for LWRs (EGATFL) in 2014. A total of 35 institutions from 14 member countries – Belgium, the Czech Republic, France, Germany, Japan, Korea, the Netherlands, Norway, Russia, Spain, Sweden, Switzerland, the United Kingdom and the United States – as well as invited technical experts from China, participated in the activities of the group.

The expert group was divided into three task forces, which addressed the following issues:

- evaluation metrics and systems assessment;
- cladding and core materials options;
- fuel options.

The efforts of the three task forces were coordinated to provide a single report, where the different contributions were jointly compiled with the objective of summarizing the existing knowledge on ATFs, including the potential benefits of each concept, and identifying the additional development needs for successful commercialization.

The report titled "State-of-the-Art Report on Light Water Reactor Accident Tolerant Fuels" was issued in 2018. The link is provided below:

https://www.oecd-nea.org/science/pubs/2018/7317-accident-tolerant-fuels-2018.pdf.

The present report reflects the consensus reached by the participating organizations. This consensus will be useful for subsequent decisions in various national and industrial programs, for example to guide technology choices and development strategies.

The objective of the report is not to prioritize the different technologies or to down select to the most promising technologies. National programs and industrial stakeholders may use the report to decide on their own set of priorities and choose the most appropriate technology based on their specific strategy, business case and deployment schedules, which vary from state to state, as well as from company to company.

2.2 HIGH-PERFORMANCE LWR FUEL DEVELOPMENT

Mechanical Properties of LWR Fuel Materials at Elevated Temperatures

Principal Investigators: Ursula Carvajal-Nunez
Collaborators: B. Maiorov

Figure 1: RUS cell used for this study. a) Cell set- up with a reference sample. We used the Rescor311 + waterglass composite that showed desired ultrasound properties but was mechanically fragile. Thin alumina wear plates were added (bonded to transducer) to improve conditions for resonance.
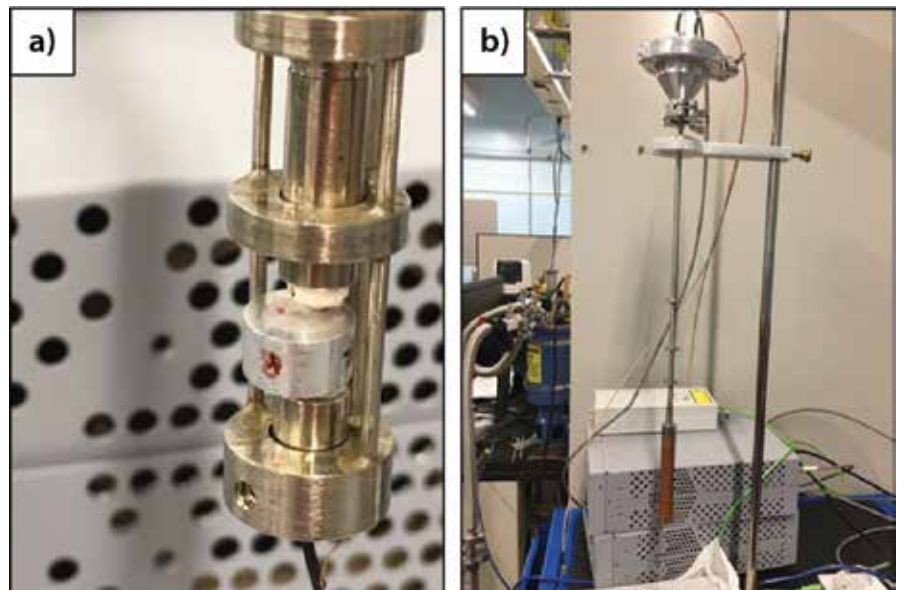
The properties of Uranium Silicides (U-Si) nuclear fuel pellets play an important role in the reactor. Evaluation of the mechanical properties to reactor relevant temperatures is necessary for fuel performance codes. Measurement of mechanical properties of conventional ceramic and intermetallic nuclear fuel materials is challenged by the fact that brittle failure dominates practical relevance at low temperatures. Resonant ultrasound spectroscopy (RUS),

a non-destructive technique, provides a demonstrated alternative to traditional mechanical test methods. The technique is useful for measurement of the elastic properties of the bulk structure of various materials. RUS is used to determine the basic mechanical properties of high density U_3Si_2 from room temperature up to 500K. The independent elastic constants C11, C44 and the bulk modulus have been evaluated to calculate the Young's Modulus and the Poisson's ratio.

This work will aim to provide baseline mechanical properties for these fuels in the unirradiated state to lay the groundwork for future extension to both higher temperatures (more relevant for nuclear reactors) and ion beam irradiations.

Project Description

The Office of Nuclear Energy (NE) currently funds work at Los Alamos National Laboratory (LANL) focused on the investigation of novel ceramic nuclear fuel concepts under the Advanced Fuels Campaign (AFC). One of the primary performance objectives targeted for improved performance of nuclear fuels is fracture toughness. New materials or composite fuel architectures that offer greater resistance to cracking under the extreme environments encountered during nuclear reactor service would provide significant improvements to steady state (e.g. heat transfer, fuel redistribution) and transient (e.g. radionuclide release at elevated temperatures) conditions. Options currently under investigation include 'nontraditional' nuclear fuels designated around high uranium density. These include uranium silicides, uranium borides, and composite fuel materials constructed of these and uranium nitride or uranium dioxide. Preliminary screening of the thermophysical and thermodynamic properties of these concepts has provided confidence in their soundness, but evaluation of their mechanical properties at relevant temperatures must be executed in order to support further study. Coupling the tools available in the Fuels Research Laboratory and at the Magnetic Laboratory, will enable synthesis and mechanical measurements of these novel nuclear materials.

Figure 2: Temperature dependence of the elastic constants a) C11, b) C44, c) Bulk modulus, B and d) Young's modulus, E for U3Si2.
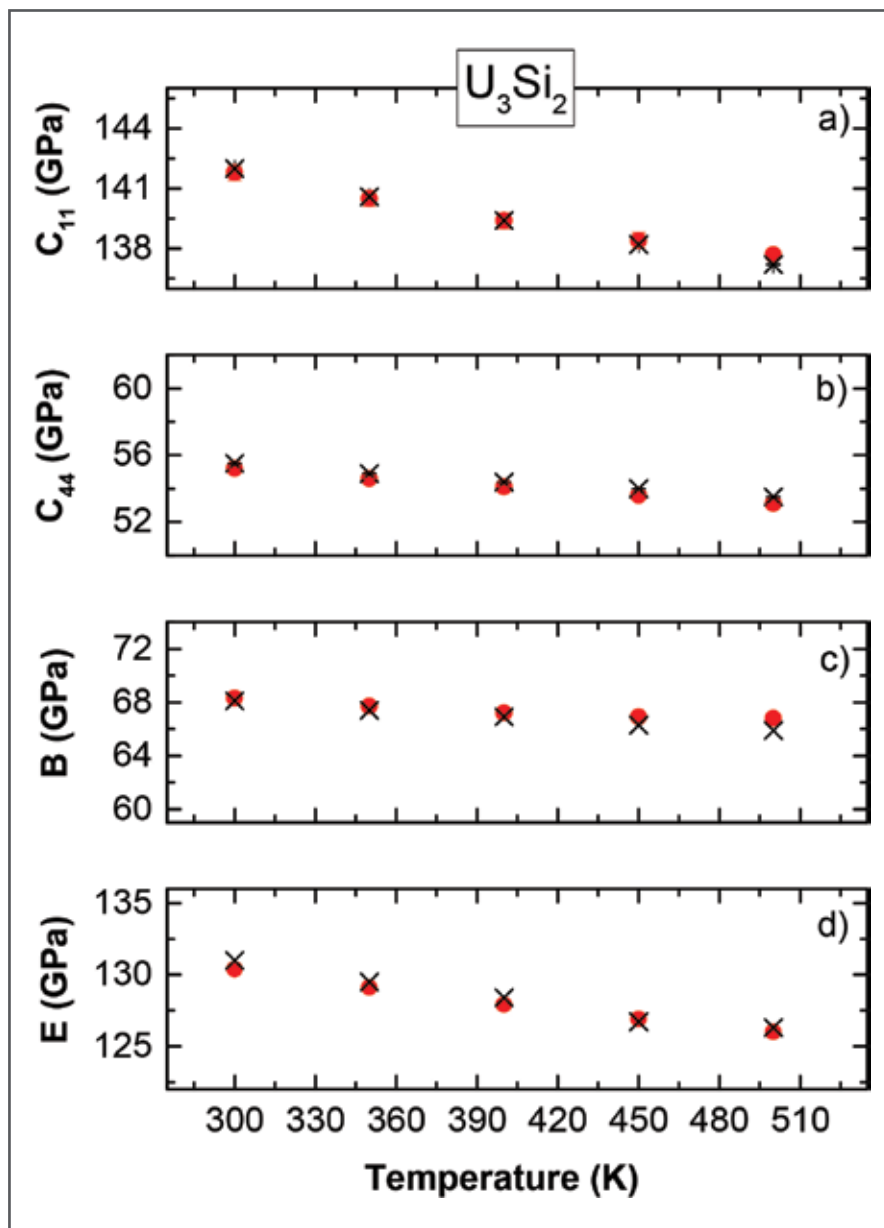
Accomplishments

Sample fabrication took place in the Fuel Research Laboratory at LANL. In order to investigate the mechanical response of the synthesized fuels. The RUS was performed from room temperature up to 500 K mechanical testing in collaboration with B. Maiorov (Magnetic Laboratory, Los Alamos National Laboratory) (Figure 1). RUS is a non-destructive technique that provides high accuracy measurements of elastic moduli. The technique is useful for measurement of the elastic properties of the bulk structure of various materials and different compositions. RUS measurement generates mechanical full body resonance spectrum. From this spectrum, an inversion scheme can be used to extract the elastic moduli.

In this study, RUS is used for the first time to determine the preliminary basic elastic properties at high temperatures of high density CeO_2, UO_2 and U_3Si_2. The independent elastic constants C11, C44 and the bulk modulus have been evaluated to calculate the Young's Modulus and the Poisson's ratio to provide data for ongoing and future studies of the fundamental behavior of these materials and conjugate mixed oxide studies. The results of this study contributed to the mechanical database for U_3Si_2 determined using RUS up to 500 K (Figure 2).

These data provide a baseline for future mechanical testing studies at higher temperatures (a new RUS cell running up to 900 K is in preparation) where irradiation or other stimuli are employed to alter the structure and chemistry of these candidate nuclear fuel materials.

Thermal conductivity of candidate ATF, burnable absorber phases: UB$_2$ and UB$_4$
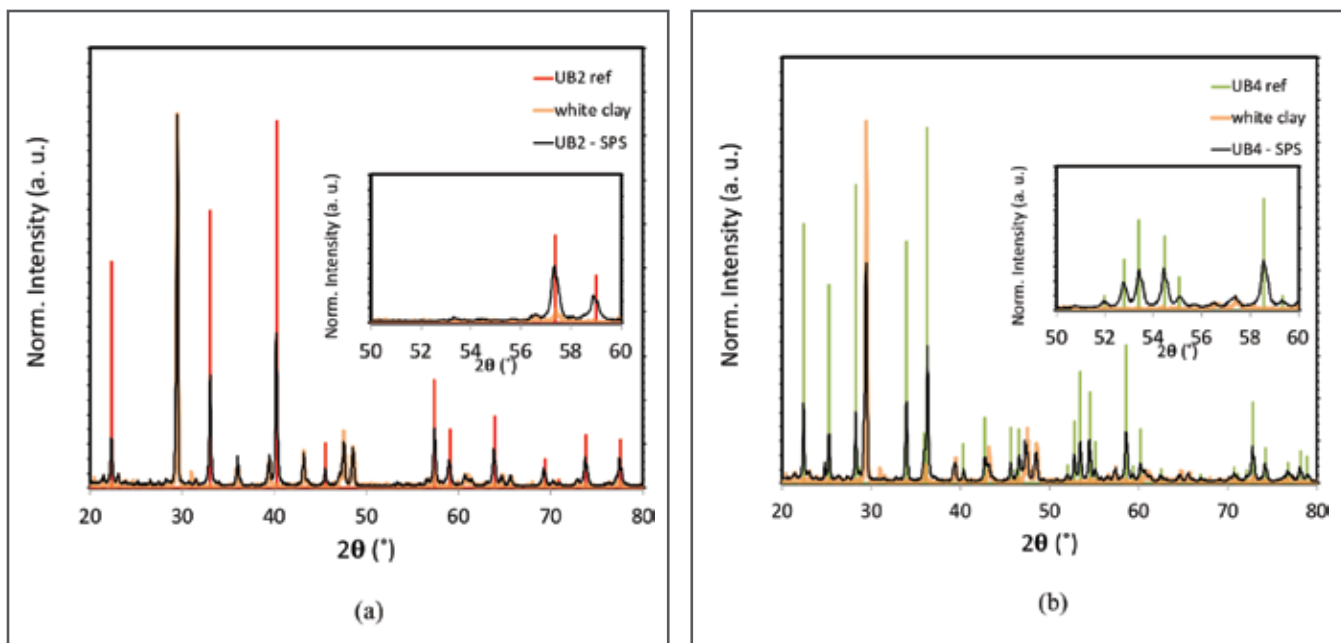
Principal Investigator: Erofili Kardoulaki

Collaborators: Darrin Byler, Ursula Carvajal Nunez, Ken McClellan – Los Alamos National Laboratory, Andy Nelson – Oak Ridge National Laboratory, iankai Yao, Bowen Gong, Jie Lian - Rensselaer Polytechnic Institute

Figure 1: Achieved densities for all UB$_2$ and UB$_4$ samples SPS sintered under a 5 minute hold, compared to conventionally sintered UB$_2$ samples under a 4 hour hold [8].
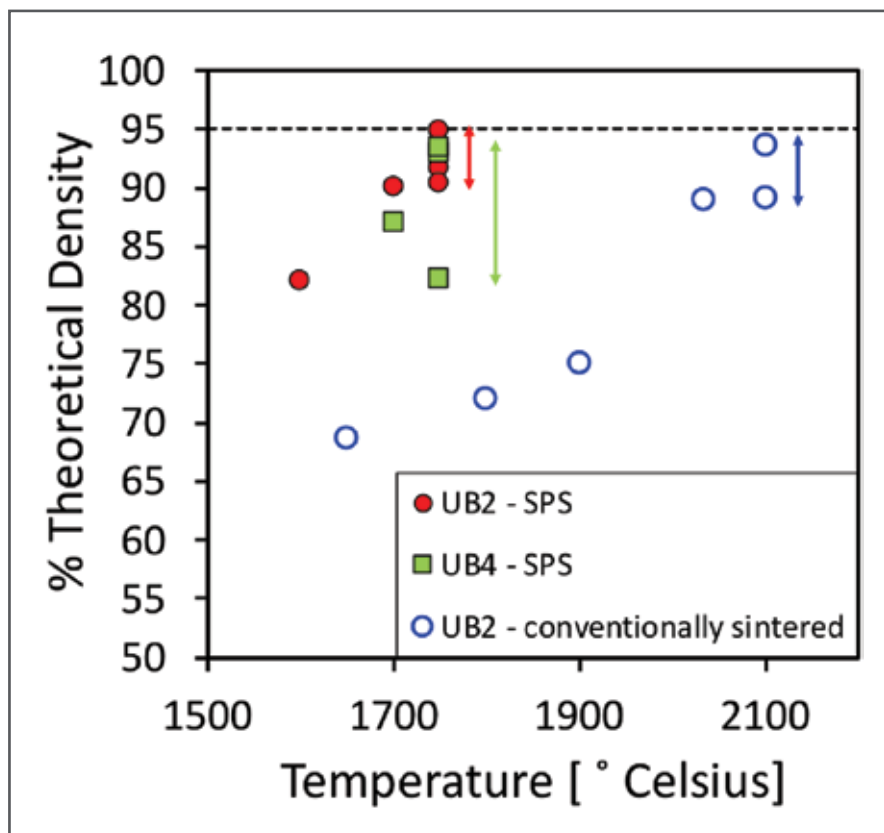
Uranium borides such as UB$_2$ and UB$_4$ have been shown to have high theoretical thermal conductivity [1] and can provide the additional advantage of acting as an efficient burnable absorber when tailoring the enrichment ratio of 10B/11B, due to the large neutron cross-section of 10B. Despite the interesting properties of uranium borides, these materials have not been extensively studied in the literature, partly due to the difficulties associated with their fabrication using conventional techniques. Field assisted sintering (FAS) techniques describe a group of novel sintering methods that use electric field and current to provide powder densification in very short time periods (minutes compared to hours) and at low temperatures compared to conventional sintering. One such technique is Spark Plasma Sintering (SPS) which uses field, pulsed direct current, and uniaxial pressure to densify

Figure 2: XRD results of (a) UB_2 and (b) UB_4 samples sintered by SPS at 1750 °C compared to reference data (UB_2 - [9], UB_4 - [10]).

powders loaded into a graphite die at short timescales. SPS has been shown to densify materials with high melting points in reduced times and temperatures [2]–[7].

Project Description:

In this study, SPS was used to densify monolithic UB_2 and UB_4 pellets to high densities, i.e. higher than 90% theoretical density (TD) for thermal conductivity measurements since these values have not been reported in the literature. This is an important first step to increase understanding in these materials and assess the potential benefits in using them as part of composite accident tolerant fuel concepts since

their thermal conductivity is expected to be significantly higher than that of UO_2. Previously White [8] had measured the thermal conductivity of conventionally sintered UB_2 but these values were measured on an 80% TD pellet and are therefore expected to be significantly lower than the measurements performed on high density pellets, fabricated by SPS. The UB_2 and UB_4 materials for this study were synthesized via arc melting and the resulting buttons were crushed, milled and sieved to produce the feedstocks used in this study. The feedstocks were then shipped to Rensselaer Polytechnic Institute (RPI) where the SPS was performed. SPS parameters

Figure 3: SEM backscatter image showing the typical microstructure of an (a) UB$_2$ and (b) UB$_4$ sample sintered by SPS at 1750˚C.
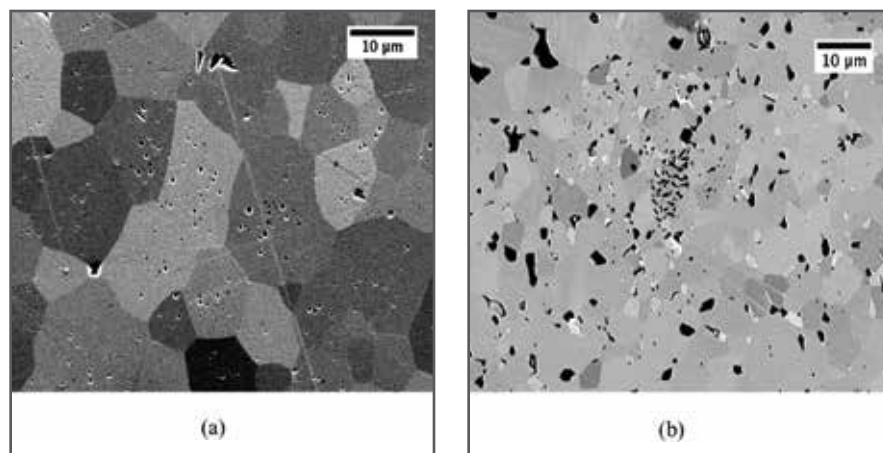
such as temperature and hold times were tested to select the optimum conditions that resulted in high density pellets. These dense UB$_2$ and UB$_4$ pellets were then examined for purity and mechanical integrity via Scanning Electron Microscopy (SEM) and X-Ray Diffraction (XRD). The pellets were then shipped back to Los Alamos National Laboratory (LANL) and were prepared for thermal diffusivity and thermal expansion measurements via Laser Flash Analysis (LFA) and dilatometry. These measured values were used along with specific heat values from the literature to calculate their thermal conductivity as a function of temperature up to 1500˚C.

Accomplishments:

The full set of tested SPS temperatures under a 5 minute hold and resulting densities, measured via immersion density, are shown in Figure 1 along with the densities reported by White [8] from conventional sintering of UB$_2$ under a 4 hour hold. To achieve densities above 90% TD for UB$_2$ samples with conventional sintering, a 350°C increase is necessary compared to the tempera-

tures required for SPS. Both UB$_2$ and UB$_4$ samples show a decrease in achieved density when tested a few weeks apart under the same conditions. This is true for samples that were sintered via SPS and conventionally, and it shows that these materials are very susceptible to oxygen impurities. Since UB$_2$ and UB$_4$ powders can pick up impurities even when kept in glovebox atmospheres with very low levels of O$_2$ it is recommended that they are prepared in small quantities to be used immediately.

The XRD patterns from the sintered pellets with the highest densities, namely those sintered at 1750˚C are shown in Figure 2 for (a) UB$_2$ and (b) UB$_4$, respectively, together with reference data for each material (UB$_2$ - [9], UB$_4$ - [10]). The pattern from the white clay that was used as a holder for both these scans is also included in Figure 2. Despite some interference from the clay pattern, both the UB$_2$ and UB$_4$ samples are shown to be phase pure as no secondary phases were detected. In Figure 3, the typical microstructures, taken in backscatter
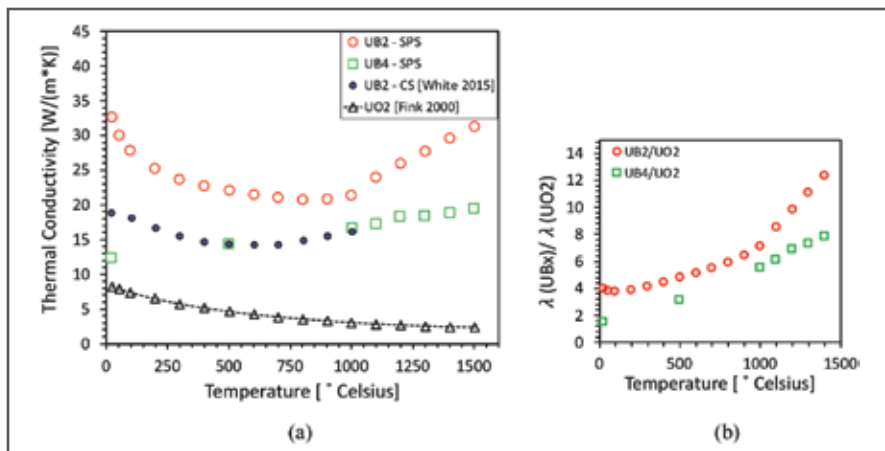
mode, from a polished (a) UB_2 and (b) UB_4 sample are shown. The UB_2 sample (95% TD) is phase pure with some intergranular porosity and average grain size of 7.5 μm. The UB_4 sample (93% TD) shows more extensive porosity, as expected, and signs of a small fraction of a secondary phase (bright phase). Since the XRD results indicate a phase pure material, this secondary phase is expected to be less than 5%. The average grain size for this sample is 4 μm.

The thermal conductivity of UB_2 and UB4 has been calculated up to 1500°C using experimental data for the thermal expansion and thermal diffusivity, and literature values for specific heat (UB_2 - [11], [12], UB_4 - [13]). It is shown in Figure 4(a) as a function of temperature along with literature data for UO_2 [14] and the conventionally sintered UB_2 pellet [8]. The calculated conductivity for UB_2 is significantly higher than that of UB_4 at room temperature and at higher temperatures, past 1000°C. The conductivity reported for the conventionally sintered pellet, although it follows the same trend as that for the SPS sample measured in this study,

it is significantly lower over the entire range of measured temperature. At temperatures past 1000°C the thermal conductivity increases for both UB_2 and UB_4, meaning at these temperatures the thermal gradients would be minimized thus making them good candidates for accident tolerant fuel. In Figure 4(b) the ratio of thermal conductivity for UB_2 and UB_4 over that of UO_2 is shown to better illustrate this point. For the case of UB_2, the conductivity is almost 12 times higher than of UO_2 at 1500°C whereas for UB_4 it is 7 times higher. The increase in the thermal conductivity for these materials is expected due to the increased electronic contributions resulting from their metallic nature. The thermal conductivity trend for UB_2 reveals that phonon contributions are significant in the lower temperatures and as the temperature increases the electronic contribution becomes prevalent while in the case of UB_4, the electronic contribution seems to be prevalent throughout the studied temperature range.

The thermal conductivity of UB_2 and UB_4 has been measured up to 1500°C and was found to increase at temperatures higher than 1000°C, thus enhancing the accident tolerance classification of these materials.

Sintering kinetics of U$_3$Si$_2$ in commercial atmospheres

Principal Investigator: Joshua T. White
Collaborators: Aditya Shivprasad and Chris Grote

Figure 1. Sintering traces for U$_3$Si$_2$ while varying the atmosphere in the glove box dilatometer. The low temperature expansion observed in the 6% H$_2$/Ar specimen is believed to be hydriding of the sample.
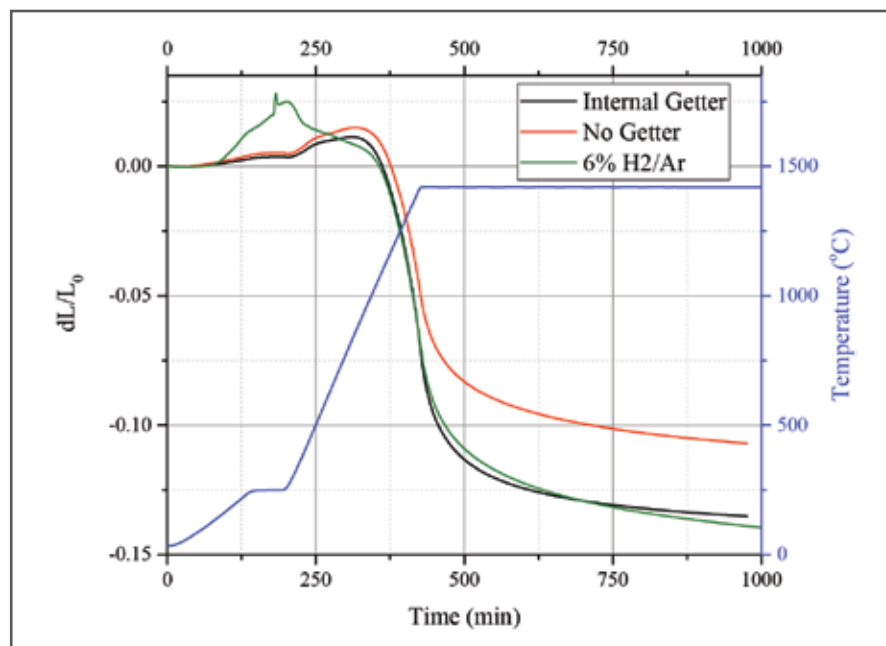
High uranium density fuels have received new interest in the Nuclear Technology Research and Development (NTRD) Program's Advanced Fuels Campaign (AFC) as a potential replacement for UO$_2$ in commercial light water reactor (LWR). Research this FY has focused on the installation of an inert atmosphere glove box with a dilatometer and simultaneous thermal analyzer. Sintering studies were then conducted on U$_3$Si$_2$ in a number of potential commercial atmospheres to study the impact this has on densification of monolithic fuel pellets.

Project Description:

One of the identified issues with high uranium density fuels is that the powders readily oxidize even when low O levels are present which leads to decreased sinterability in the powders. For example, U$_3$Si$_2$

powder that has been size reduced and exposed to air or prolonged exposure (e.g. 2 weeks) in a glove box atmosphere will decrease the theoretical density (TD) of the pellets from 98% to 79% of the theoretical density. This effect has been difficult to study given that benchtop techniques (i.e. dilatometry) expose oxygen sensitive green pellets to air prior to the experiment, which convolutes the analysis of sintering kinetics. In order to overcome this technical challenge, a dilatometer was installed within an inert atmosphere glove box to allow transfer of the samples to the instrument without exposure to oxygen. In doing so, this study provides fundamental sintering kinetic data that can be used by industrial teams to efficiently develop furnace schedules while scaling up fuel fabrication of this class of fuels.

Accomplishments:

The impact of atmosphere on the sintering kinetics of U_3Si_2 was investigated for the first time using a dilatometer installed within an inert glove box. A glove box was installed at the Fuels Research Laboratory in Los Alamos National Laboratory (LANL) with a high temperature (<1600°C) SiC furnace for the sintering studies. The inert atmosphere glove box is maintained below 0.1 ppm O_2 contamination with an Ar atmosphere to minimize the oxidation of the green pellets prior to the sintering experiment. Sintering traces using standard U_3Si_2 furnace profiles are shown in Figure 1, where the gas was varied between Ar, internal Zr gettered Ar, and 6% H_2/Ar. The internal Zr getter utilized Zr wire pieces ~2 mm from the pellet inside the dilatometer. Controlling and removing the oxygen levels within the dilatometer proved to be a technical challenge as industrial supplied gas (ultra-high purity Ar and 6% H_2/Ar) typically contains 0.3 ppm O_2 contamination. Under these conditions, sintering of U_3Si_2 was shown to have slightly decreased final densities, which was not improved with an internal Zr getter within the furnace.

Figure 2. Microstructure of U3Si2 after sintering in the dilatometer under a gettered Ar atmosphere. The interior of the pellet is shown in (a) while the pellet surface is shown in (b).
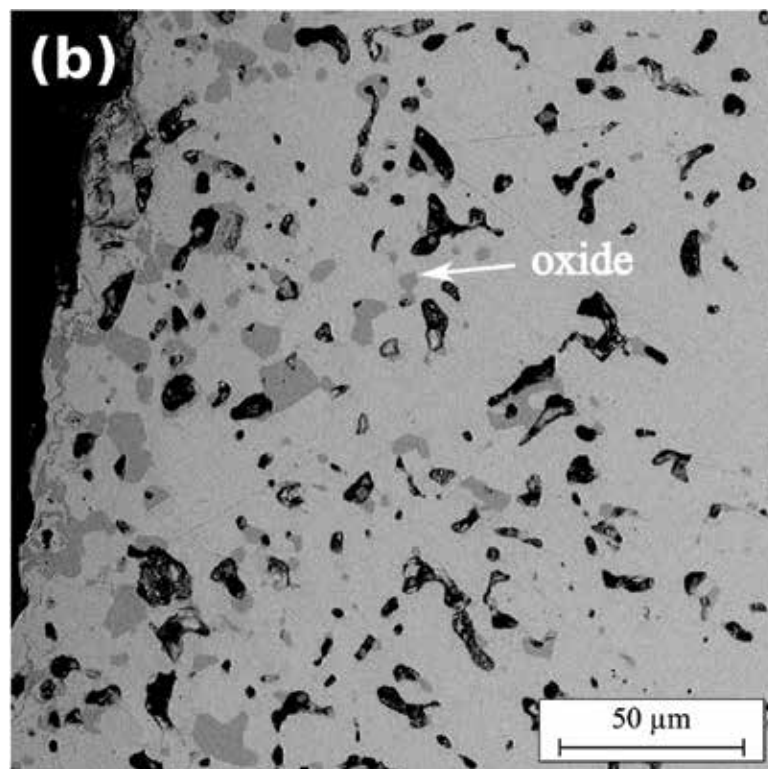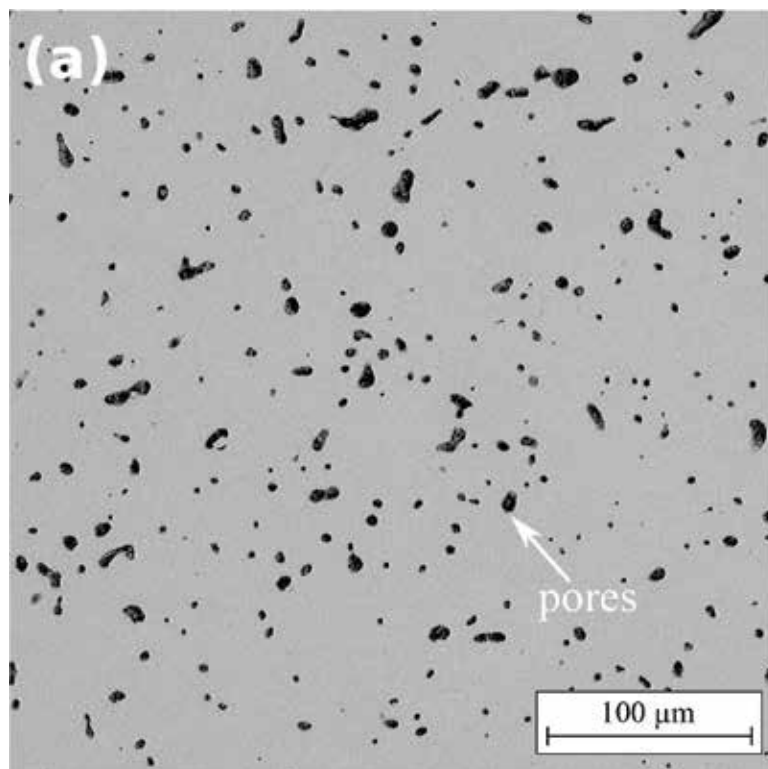
Sintering kinetic data will assist scale-up manufacturing of high uranium density accident tolerant fuels to an industrial setting.

Microstructures of the as sintered pellets are shown in Figure 2 (a), which exhibited approximately 9% porosity, with darker grey uranium oxide impurities. The surface (Figure 2 (b)) exhibited slightly more oxide contaminant. At low temperatures, it is believed that the U_3Si_2 low density green pellet allows exposure of oxygen to the powder surfaces, which forms oxides precipitates and reduces sinterability of the powder. This effect is not mitigated by the Zr getter until the oxidation of Zr is activated above ~700°C. Because of this, sintering of U_3Si_2 in the glove box dilatometer achieved around 91% of the theoretical density, slightly lower than a pristine sintering environment. Furnace ramp rate did not largely impact the resultant pellet density in this study, although further investigations are required to evaluate if the micro-structure is impacted at the higher ramp rates. Sintering in hydrogen was shown to be detrimental at temperatures between 200°C and 400°C, which is suspected to be caused by hydride formation within the green pellet.

Corrosion and hydriding behavior of U$_3$Si$_2$

Principal Investigator: Aditya P. Shivprasad
Team Member/Collaborator Information: Joseph R. Wermer and Joshua T. White

> **Studying the mechanisms of steam oxidation and hydrogen absorption will help understand how to mitigate degradation in-reactor by these modes.**

U$_3$Si$_2$ is a promising, high-uranium-density fuel because of its good thermal properties and performance during irradiation and has been considered as a potential replacement for UO$_2$ in commercial light water reactors. Despite these advantages, limitations are observed when the coolant comes in contact with the fuel, e.g. breach of cladding, resulting in degradation of the fuel. Research this FY focused on continuing research from FY17 by studying the hydrogen absorption of U$_3$Si$_2$ and relating this behavior to corrosion performance in steam.

Project description

Under steam and simulated pressurized water reactor (PWR) conditions, the fuel has also been observed to oxidize and potentially absorb hydrogen, after which the fuel pulverizes. Rapid pulverization of U$_3$Si$_2$ has been observed during exposure to both hydrogen gas mixtures, high-temperature water, and steam but is not well-understood: comparison with waterside corrosion in uranium metal showed similarities in post-corrosion microstructure, leading to the hypothesis that hydride formation occurs in U$_3$Si$_2$. Previous work in FY17 evaluated U$_3$Si$_2$ performance in hydrogenated water (5 ppm H$_2$ at 300°C) and showed that, beyond 30 days of exposure, rapid pulverization occurred. A similar effect was observed in pure hydrogen and both were correlated to a hypothesized hydride phase observed in the microstructure.

In FY18, the oxidation performance of U$_3$Si$_2$ in steam was evaluated using carefully-selected steam compositions and temperatures. In parallel, hydriding of U$_3$Si$_2$ was studied in a Sievert's apparatus at various temperatures. The purpose of these tests was to determine the temperature regimes in which hydriding kinetics and thermodynamics were sufficient to cause pulverization of U$_3$Si$_2$, as well as to characterize the composition of the hydride phase that forms.

Accomplishments

Sintered compacts of U$_3$Si$_2$ were subjected to oxidation in a 75% steam/argon mixture. Using a steam furnace coupled with thermogravimetric analysis at the Fuels Research Laboratory at Los Alamos National Laboratory (LANL). Samples were heated in argon gas maintained at
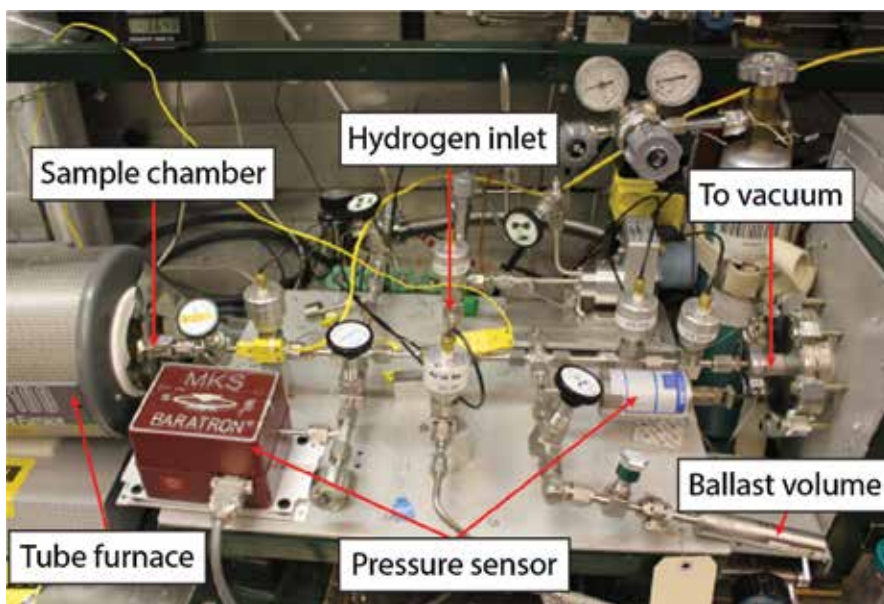
oxygen levels below 0.1 ppm; once the desired temperature was reached, the steam/argon mixture was flowed through the system and sample mass change as a function of temperature and time was recorded. Results showed that below 350°C, steam oxidation does not represent a significant degradation mode, as mass gain under these conditions was negligible and pellet integrity was maintained (no spallation of material). This is consistent with previous work at LANL under similar conditions.

Sieverts' gas absorption measurements were simultaneously performed on sintered compacts of U_3Si_2 in pure hydrogen at the Sigma Division of LANL. Samples were loaded into a steel reaction chamber in an inert glovebox. The reaction chamber was sealed, connected to a gas manifold of known volume that comprised the Sieverts' apparatus, and then evacuated. An annotated image of the experimental setup is shown in Figure 1.
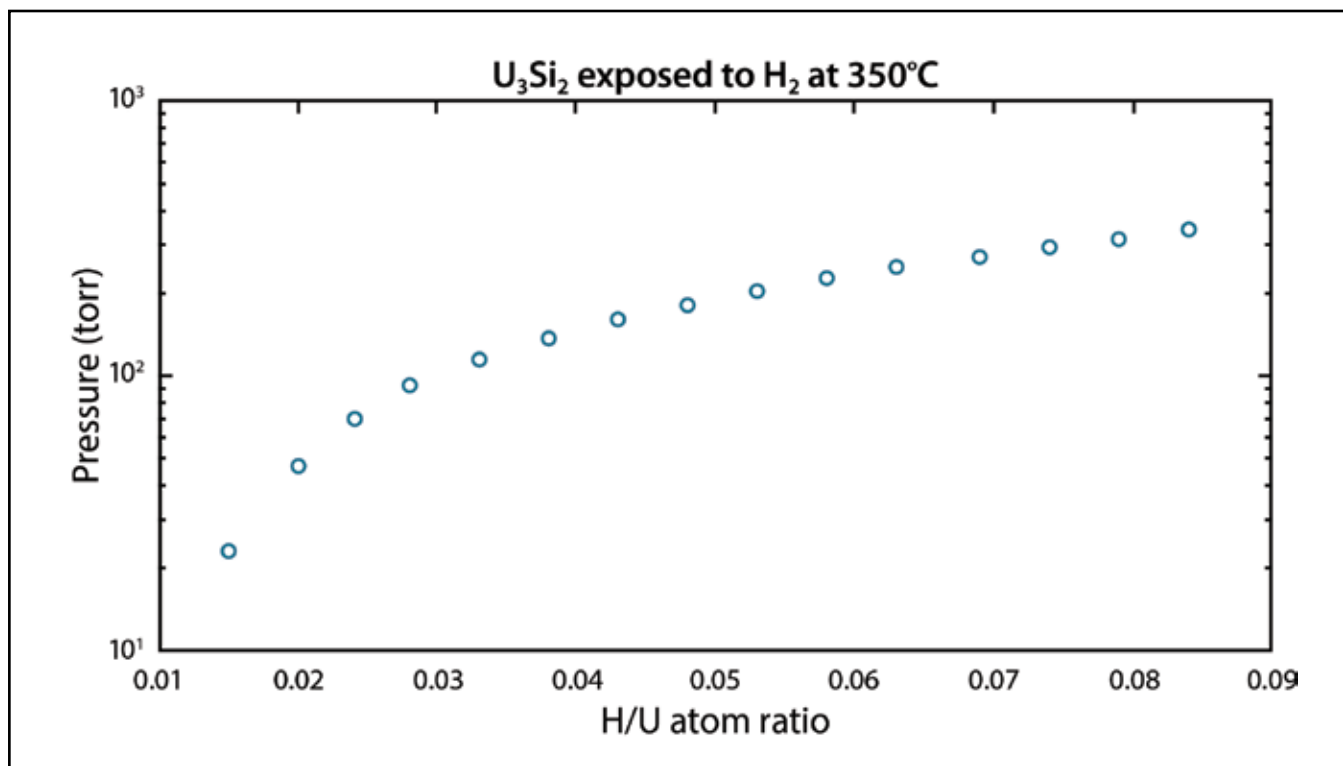
U$_3$Si$_2$ exposed to H$_2$ at 350°C

Figure 2: Pressure-composition-temperature isotherm for hydrogen absorption of U$_3$Si$_2$ at 350°C. Hydriding was terminated at 320 torr of H$_2$ and absorbed hydrogen corresponding to a H/U ratio of 0.084.

The sample reaction chamber was placed inside a tube furnace and brought to the desired temperature. Once at temperature, the manifold was filled with a known pressure of hydrogen. After the hydrogen had equilibrated in the manifold, the sample chamber was opened to the hydrogen. The amount of hydrogen absorbed by the material was measured by the change in hydrogen pressure through the ideal gas law. Pressure was measured using an MKS Baratron capacitance manometer. Incremental aliquots of hydrogen were provided to the manifold after pressure stabilization was achieved. Aliquoting was performed until the hydrogen pressure in the sample chamber reached a value of 320 torr, at which point the measurement was terminated.

Measurements were made at 250°C and 350°C using an automated method in LabView and done manually at 400°C. Hydrogen pressure was plotted as a function of the hydrogen-to-uranium (H/U) ratio calculated from the moles of hydrogen absorbed. An example plot of such a pressure-composition isotherm for 350°C is given in Figure 2, which shows that the amount of hydrogen absorbed corresponded to an H/U ratio of 0.084. At 250°C, the amount of hydrogen absorbed by U_3Si_2 was below the detectable limit and is consistent with observations in the steam corrosion at the same temperature, where pellet integrity was maintained and mass gain was minimal. At 400°C, this value increases to 0.9. Under similar temperature and hydrogen pressure, uranium metal will react with hydrogen to form UH_3, which has an H/U ratio of 3. Because of this discrepancy, it is hypothesized that the hydride phase that forms is not UH_3, though disproportionation of U_3Si_2 by hydrogen to form UH_3 and Si cannot be ruled out. In this case, Si would only increase the barrier for absorption of hydrogen, requiring higher hydrogen pressures to form UH_3.

How the hydrogen absorption reaction of U_3Si_2 proceeds is still not yet well understood, nor is the regime wherein this phase significantly contributes to degradation of U_3Si_2 in steam. To that end, ongoing work will attempt to examine the structure of the hydride phase and temperature above which the hydride phase is no longer stable in steam conditions. Additionally, because the hydrogen absorption reaction did not appear to terminate, future work will focus on higher pressures of hydrogen and comparison with uranium metal under the same conditions.

Oxidation Behavior and Thermophysical Properties of UN/UO$_2$ Fissile Composite

Principal Investigator: Nicholas R. Wozniak
Collaborators: Joshua T. White

Figure 1. Thermograms comparing the mass gain of the composite materials and the monolithic UN and UO$_2$ during ramped heating to 1000°C under 62% to 83% H$_2$O (g) atmosphere.
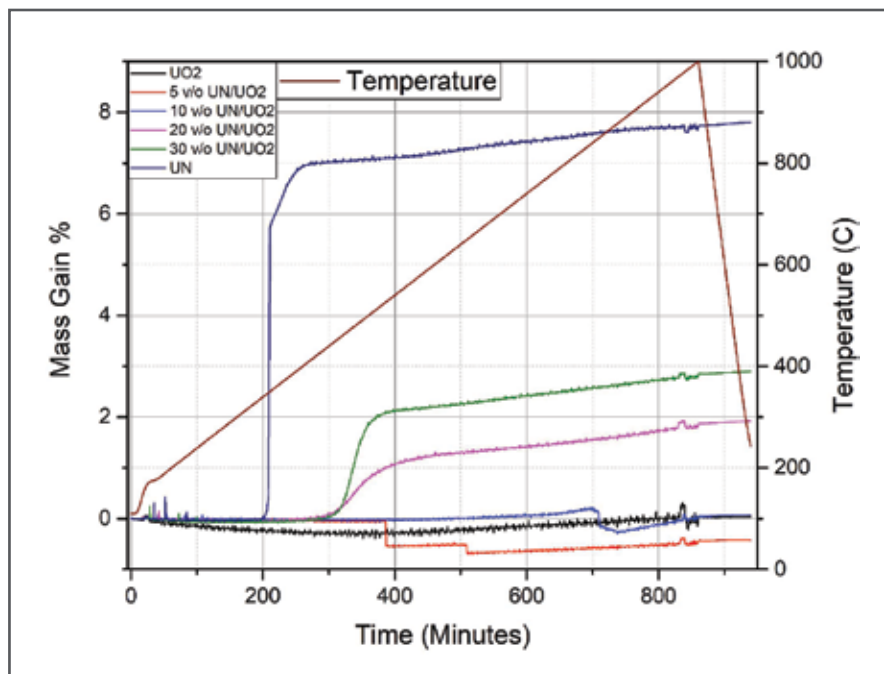
Composite uranium materials could be used as a fuel alternative that would combine advantages of phase pure fuel forms. Traditionally used UO$_2$, while thermodynamically stable, has a less than desirable thermal conductivity. Uranium nitride fuels would have a higher uranium atom density, a higher melting temperature, and a higher thermal conductivity, albeit with poor oxidation resistance. A composite fuel consisting of uranium nitride (UN) in a UO$_2$ matrix, could exhibit the strong oxidation resistance of UO$_2$, while increasing the uranium atom density and thermal conductivity.

Project Description:
Composite fuels consisting of UN in a uranium oxide matrix could have significant safety advantages over either material in a monolithic form. The increased thermal conductivity, compared to tradition UO$_2$ fuels, would result in a decreased temperature gradient across the fuel and subsequently less cracking of the

pellet. The UN additions would also increase the uranium atom density, improving the neutron economics of the fuel. The UO_2 matrix would improve the oxidation behavior, compared to UN fuels, minimizing washout in accident scenarios.

The oxidation behavior of the UN/ UO_2 composites was studied by two different techniques; oxidation in simultaneous thermal analyzer equipped with a water vapor furnace and water vapor generator and in autoclaves. These test conditions were chosen to mimic possible Light Water Reactor (LWR) accident conditions, such as a loss of coolant or leaker accident. Testing of dense composite compacts in the simultaneous thermal analyzer was performed under ambient pressure and consisted of isotherms at 350°C and thermal ramps to 1000°C under steam atmospheres ranging from 62% H_2O (g) to 82% H2O (g), depending on temperature. The steam atmospheres were chosen such that the UO_2 would not oxidize past UO_2.25 and would remain in the fluorite phase field. The autoclave testing was performed on dense composite compacts at 300°C and 86 bar, for durations of either 2,5, 10, or 20 days. Solid state Ni/NiO buffers were used to maintain hydrogen levels in the 1-5 ppm range to simulate LWR coolant chemistry.

Accomplishments:

Composite materials have been synthesized and the thermophysical, oxidation behaviors, and microstructure have been explored. Composites compacts consisting 5, 10, 20, and 30 volume percent UN in UO_2 have been produced with densities approximately 90% theoretical density (TD). The oxidation behavior has been explored in air, under steam atmospheres at ambient pressure, and under pressurized water conditions.

Dense compacts of UO_2, UN, and the four composite materials were exposed to steam at an isothermal temperature of 350°C for 12 hours to mimic a leaker type scenario in an LWR. The expected O/U ratio of UO_2 at these conditions was greater than 2.25 indicating oxidation of the material to higher oxide phases, leaving the fluorite structure of UO_2+x, leading to pulverization of the pellet and material washout. Of the six pellets tested, only the monolithic UO_2 and the 5 volume percent composite were not pulverized under these conditions. The calculated O/U ratios of the pellets following oxidation were lower than expected, suggesting incomplete oxidation of the UN to oxide phases. This was supported by the presence of UN in the diffraction pattern.

The six materials were also tested by thermally ramping from 150°C to 1000°C under a steam atmosphere. The thermograms for this testing are shown in Figure 1. The large sudden drops in mass gain are due to material ejection from the crucible. These material ejections are only seen in the 5 and 10 volume percent composite pellets, but the pellets did not completely pulverize. The higher UN containing composites and the monolithic UN pellets did pulverize under these conditions, as seen in Figure 2. Similarly to the 350°C isotherm, the calculated O/U ratios following oxidation were lower than expected, but to a lesser extent, suggesting incomplete oxidation of UN. The oxidation onset is delayed in temperature for the composite materials compared to monolithic UN and is much more gradual in rate. These ramps show some protection given to UN from the UO_2 matrix. This is supported by the higher oxidation onset temperature of the composite material compared to monolithic UN and the presence of UN in the resulting diffraction patterns. The lower UN composite materials did not show significant oxidation and the pellets did not pulverize. Loss of material due to ejection is still under investigation, but similar features have been observed in other high U-density fuels under similar conditions.

Each of the four composite materials were tested in autoclave conditions at 300°C and pressures of 86 bar for either 2, 5, 10, or 20 days. Under these conditions, UO_2 is expected to

oxidize to O/U ratios greater than 2.25, leading to pulverization and washout. Following autoclave testing, the pellets were visually inspected for pulverization and a mass obtained, where possible. It was not possible to determine O/U ratios due to material that was lost to the autoclave. The 5 volume percent pellets did not pulverize at any of the times tested, and did not show any appreciable mass change. Ten volume percent pellets did not pulverize up to 10 days. All 5 and 10 volume percent pellets (with the exception of the anomalous 20-day 10 volume percent pellet) had diffraction patterns consisting only of UO_2. The 20 and 30 volume percent pellets pulverized at all times tested and the resulting powder consisted of UN, UO_2, and higher oxide phases. This was consistent with the isothermal data taken at 350°C for 12 hours, and again further suggest oxidation protection of UN from UO_2 for the lower UN composites (5 and 10 vol. %).

Using a composite fuel form consisting of uranium nitride in a uranium oxide matrix shows promise for increasing thermal conductivity and uranium atom density, compared to UO_2, while increasing the oxidation resistance of monolithic UN. The results here indicate that further studies are needed to determine the amount of UN that can be incorporated within a UO_2 matrix, before having detrimental effects on the oxidation behavior.
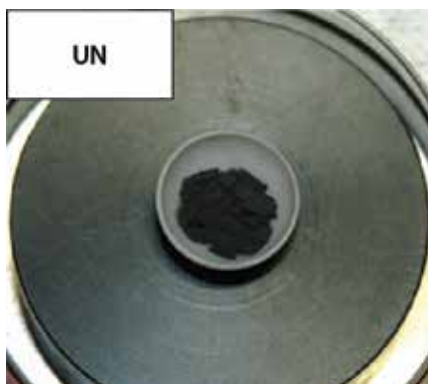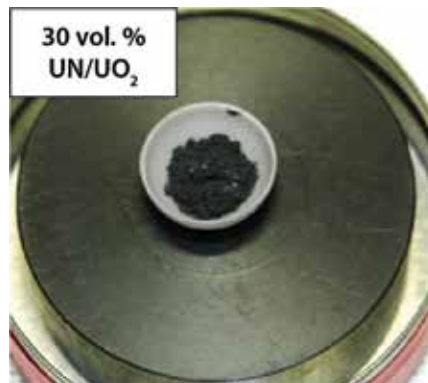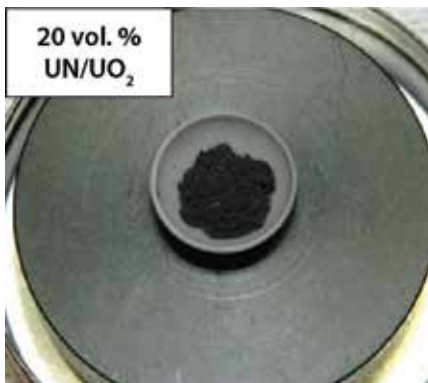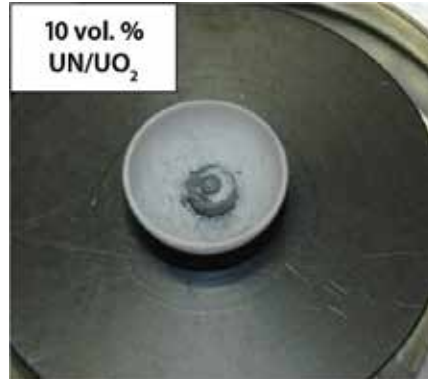
Figure 2. Photographs of pellets following thermal ramps under H_2O (g) atmosphere.

Fabrication of High Density UN Kernels for Fully Ceramic Matrix Fuels

Principal Investigator: J. McMurray
Collaborators: K. Terrani

> **Gadolinium can be added as a burnable poison to UN derived from the sol-gel method while at the same time increasing theoretical density.**

Uranium nitride (UN) has a high U density that is an important property for the accident tolerant fully ceramic microencapsulated fuel design. Gadolinium is burnable poison important for flattening the reactivity of the fuel.

Project Description:

A technological challenge associated with use of UN as the kernel for fully ceramic microencapsulated fuel, a candidate accident tolerant fuel form, is the reactivity must be decreased at the beginning of life with a burnable poison such as Gd. The focus of this research was to explore the viability of Gd additions to the sol-gel feedstock used to produce UN microsphere. An important physical property is the density, therefore the final density after carbothermic reduction and nitriding (CTRN) was measured and compared to benchmark UN. The impact of the chemical form of Gd additions to the sol-gel feedstock was investigated. It was shown that Gd is viable as an additive to UN microspheres and that very high theoretical densities (>95%) can be attained when introduced as a nitrate hexahydrate to the sol-gel broth

Accomplishments:

The goal of the research was to investigate the viability of Gd additions to UN microspheres. A significant discovery was that samples with Gd added using gadolinium nitrate hexahydrate, $Gd(NO_3)36H_2O$ showed clear increases in density over those without it prepared with the same carbon, dispersant and processing parameters. Gadolinium mononitride is isostructural with UN; it was anticipated for the added Gd to be incorporated into the UN matrix to form (U,Gd)N. One aim of this work was to determine if solid solution sintering densification mechanisms would be active. Since the X-Ray Diffraction (XRD) results for the sample with 0.125 Gd by metal fraction showed ~12.5 wt% Gd_2O_3, it is not known whether the densification mechanism results from the solid solution, precipitation, or some combination thereof.

The density decreased significantly over the baseline for Gd added as an oxide in the form of Gd_2O_3. It is proposed that Gd is better mixed when added as $Gd(NO_3)36H_2O$ which dissolves in the ADUN whereas Gd_2O_3 nanoparticles form a suspension. Further, the dispersant effect could be altered by Gd_2O_3 resulting in C agglomerations that are thought to be detrimental to achieving high density kernels.

While the Gd added as $Gd(NO_3)36H_2O$ has been shown to improve the overall density of UN, it also has the effect of displacing U. It is reported that, 12.8 g/cm^3 is required for the FCM design. Using the highest density numbers from this study only 12.4 g/cm^3 was attained. Other routes must be taken to achieve higher U densities, for example hot isostatic pressing which has been shown to produce up UN microspheres with up to 13.1 g/cm^3.
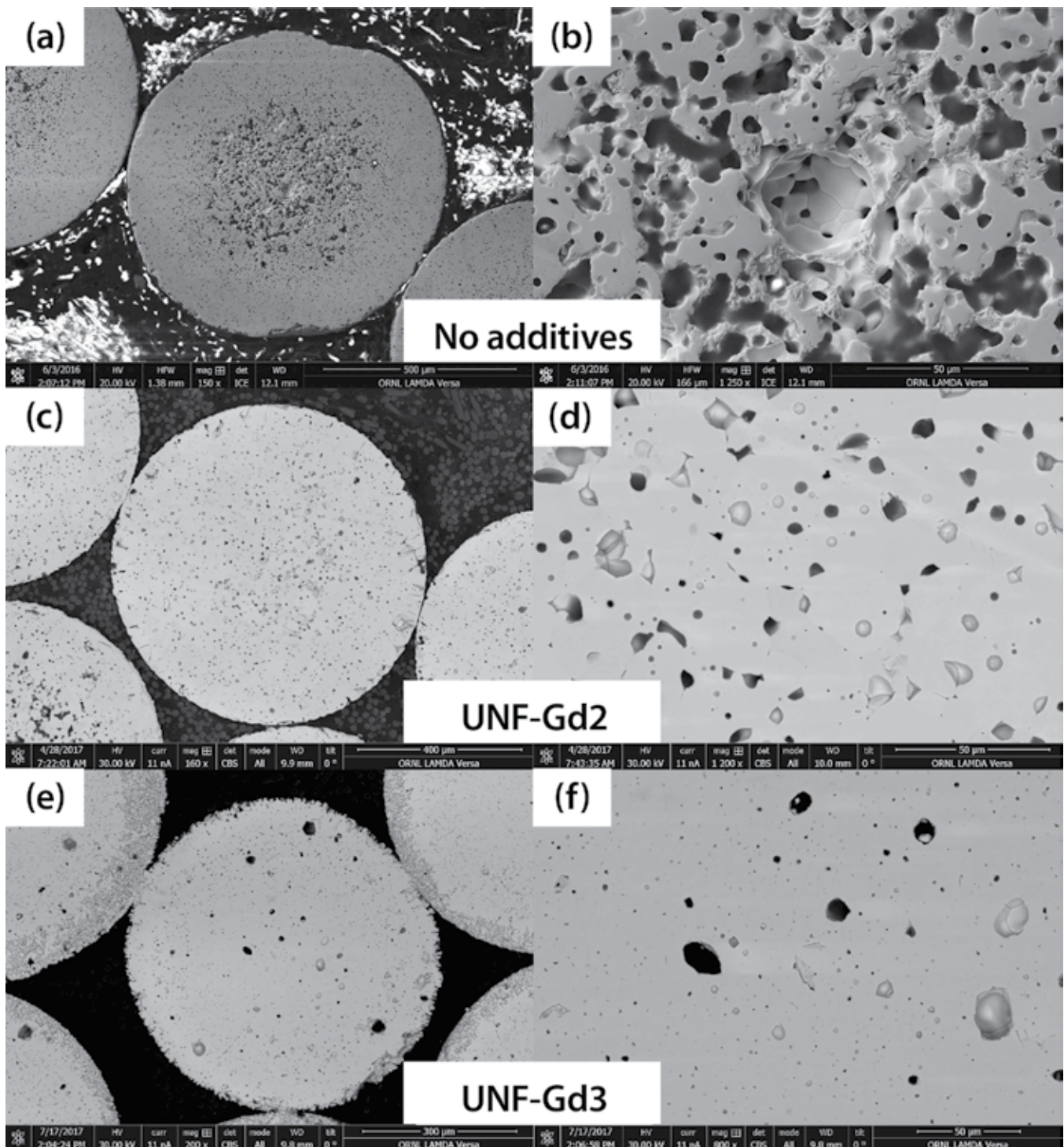
Figure 1. Typical microstructures of UN with no Gd (a-b), UN with Gd 0.025 heavy metal mole fraction (c-d), and UN with Gd 0.125 heavy metal mole fraction (e-f). High-resolution micrographs in each case was obtained focusing on the center of the samples.

The Grain-size Effect on Thermal Conductivity of Uranium Dioxide

Principal Investigator: Krzysztof Gofryk (INL)
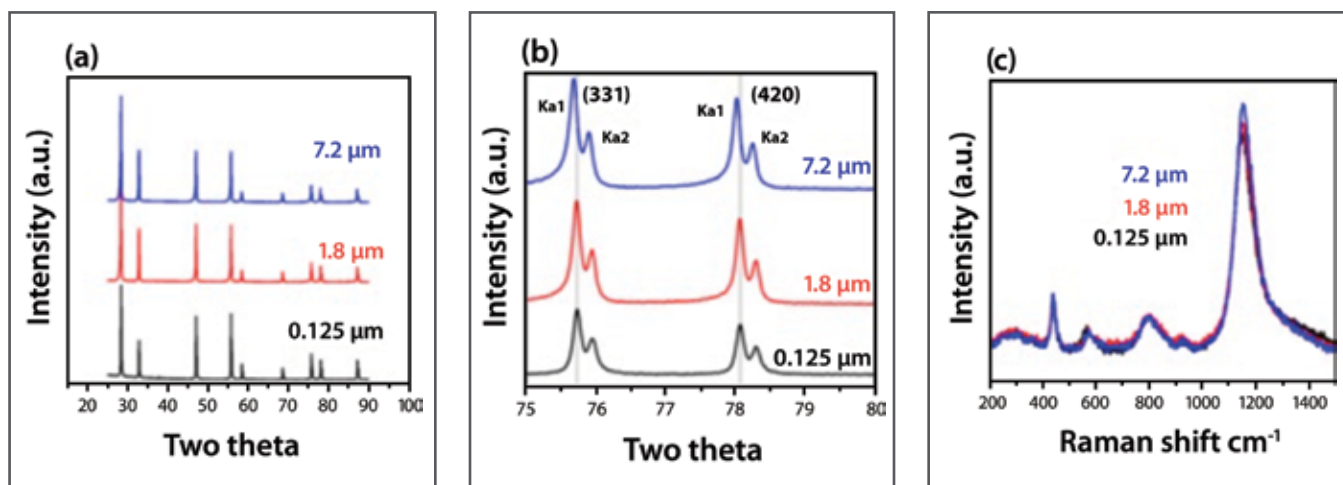*Collaborators: Jie Lien (RPI), Michael Tonks (UF)*

Figure 1. (a) XRD spectra show that the sintered pellets have UO_{2+x} structure with `x' values calculated by peak positions as shown in (b) for the high angle section. Superimposing feature of Raman spectra (c) indicates similar degree of interaction between defects and UO_2 crystal structure in the sintered pellets.

Uranium dioxide is one of the most studied nuclear materials as it is used as the primary fuel in the commercial nuclear reactors. There are around 500 active nuclear reactors, producing more than 15% of the total electricity worldwide. In a reactor, the heat energy produced from the nuclear fission events inside the fuel pellets is transformed into electricity. Thus, the heat transport mechanism, i.e., thermal conductivity of the fuel material is an important parameter for fuel performance, regarding its efficiency and safety.

Project Description:
A nuclear reactor operates at extreme environments that can include high temperature, high pressure, and high irradiation. As a result, a fuel pellet undergoes severe structural changes under irradiation conditions, including grain subdivision, fission gas bubbles growth and redistribution and extended defects accumulations. Thermal properties of the fuel

material are greatly affected by these changes which ultimately affect the performance of a reactor. Numerous theoretical and experimental studies have been carried out to understand how these microstructure changes affect thermal transport properties of UO_2. In an oxide fuel, the lattice vibrations (phonons) responsible for the heat transport are scattered by different scattering centers, such as defects, grain boundaries, and phonon-phonon interactions. Depending upon the temperature range, different scattering mechanisms dominate at different temperature regimes. For instance, umklapp phonon-phonon scattering dominates the thermal conductivity at high-temperature, while the point-defect and boundary scattering govern the heat transport at intermediate and low temperatures,

respectively. At low temperatures where the phonon mean free path is comparable to the grain-size, the grain boundary scattering mechanism is the main factor limiting the thermal conductivity. Here we have investigated the grain boundary scattering effect on the thermal transport behavior of uranium dioxide having different grain-sizes. We analyzed the results obtained using molecular dynamics (MD) calculations.

Accomplishments:

We have carried out systematic studies on the grain-size effect on thermal conductivity of UO_2 by performing measurements at low temperatures to study different scattering mechanisms, focusing on grain boundary scattering. The polycrystalline samples having
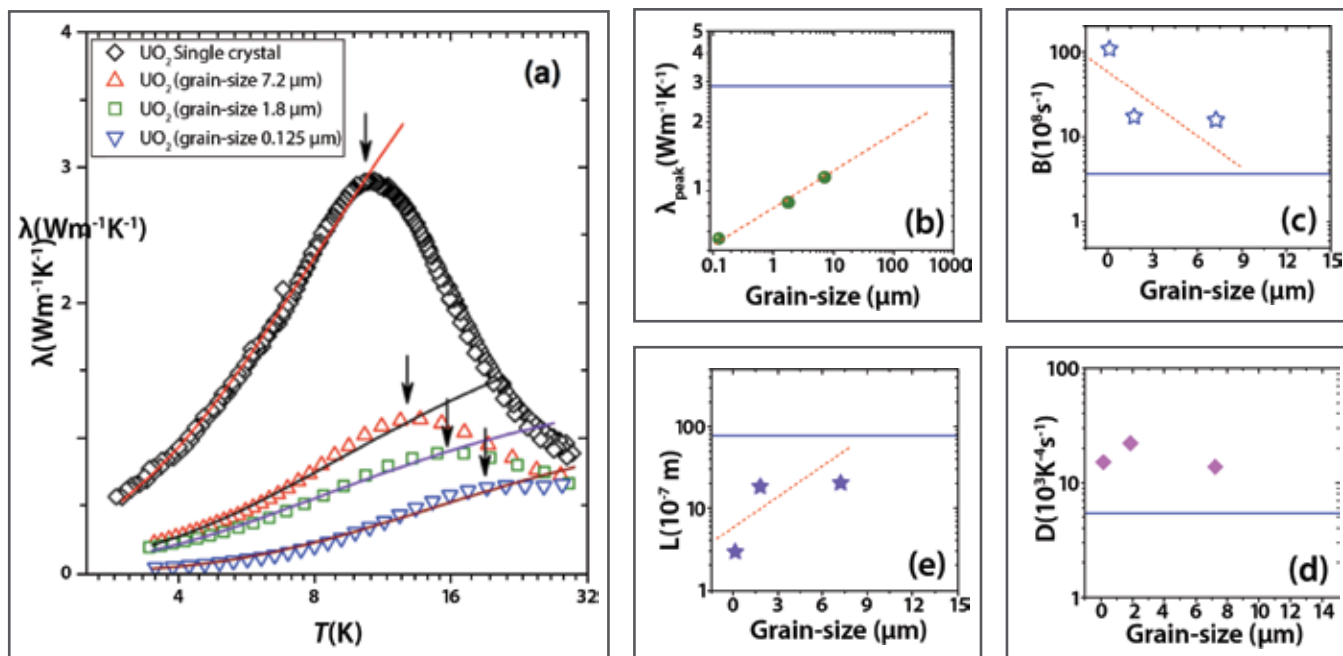
Figure 2. (a) The low temperature thermal conductivity of UO₂ samples. The solid lines represent the least-square fits of the Callaway model to the experimental data; The grain-size dependence (obtained from the Callaway model) of the low-temperature thermal conductivity peak (b), parameters B (c), D (d), and the mean free path of phonons L (e). The corresponding values for the UO₂ single-crystal are displayed as a horizontal line in the relevant graphs. The dashed lines are guides to the eye.

different grain-sizes (0.125, 1.8, and 7.2 um) have been prepared by spark plasma sintering technique at Rensselaer Polytechnic Institute(RPI) by Prof. Jie Lien research team. After the synthesis several characterization techniques have been employed such as by x-ray powder diffraction (XRD), scanning electron microscope (SEM), and Raman spectroscopy (see Fig1) to ensure sample quality with desired

characteristics. All pellets were fully densified with measured density higher than 95 % TD. The XRD spectrum in Figs. 1a and b show that the sintered pellets are single phase UO_2. Microstructure characterization was conducted by SEM microscopy and the average grain sizes were estimated to be 0.125, 1.8, and 7.2 um. Furthermore, the Raman spectroscopy (Fig.1c) shows the chemical bonding in those three-different grain-sized samples are

very similar, indicating comparable localized defect interaction with the crystal structure of UO_2. The thermal transport properties (the thermal conductivity and thermoelectric power) have been measured in the temperature range 2-300 K and the results were analyzed in terms of various physical parameters contributing to the thermal conductivity in these materials in relation to grain-size. These studies have by performed at Idaho National Laboratory (INL) by Dr. Gofryk's team. We show that the grain boundary scattering parameters vary systematically with the grain-size below 30 K (Fig.2a). Such a behavior is not observed at higher temperatures where other scattering processes start to dominate. At room temperature the thermal conductivity values of the UO_2 polycrystalline samples vary slightly with grain-size (not shown here), and all the values are within 5% error of the measurement (and also ~10% smaller than that of the single-crystal). The variation of thermopower (not shown here) suggests that very small oxygen off-stoichiometry might be present and play a role in lowering of the

thermal conductivity in the UO_2 samples, especially at high temperatures. To evaluate this important observation in more details, single crystals studies (on UO_2 crystals with different oxygen content) are required. The thermal conductivity data were analyzed and the variation of different parameters such as grain boundary (B), defects (D), or phonon mean-free path (L) with the grain-size are presented in Figs 2b-e. In order to determine if changes in the measured thermal conductivity at high temperatures are due to the grain boundary Kapitza resistance, Prof. Michael Tonk's team (University of Florida) investigated this behavior using the analytical Kapitza resistance equation. All the results obtained indicate importance of low temperature measurements to study grain boundary scattering in UO_2. At higher temperatures other scattering processes are involved in the heat transport in these materials that makes this analysis difficult. In addition, these studies point to importance of oxygen off-stoichiometry for the thermal conductivity of UO_2.
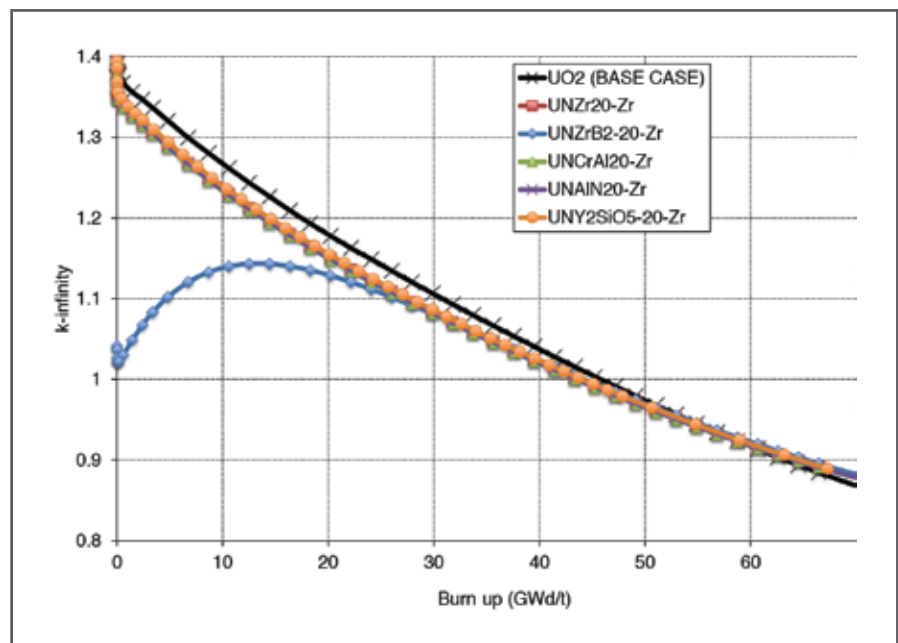
Initial Screening of Impacts on Reactor Performance and Safety Characteristics of Advanced Fuels for LWRs

Principal Investigator: M. Todosow
Collaborators: L-Y Cheng, A. Cuadra

Figure 1. Multiplication Factor vs. Burnup (GWd/t) for UN Fuel Pellets with 20 μm Coatings

In the aftermath of Fukushima, a focus of the Department of Energy Nuclear Energy (DOE-NE) Advanced Fuels Campaign (AFC) has been the development of advanced nuclear fuel and cladding options with the potential for improved performance in an accident; these are referred to as Accident Tolerant Fuels (ATF). An assessment of the impacts of advanced Light Water Reactor (LWR) fuels/cladding on reactor performance and safety characteristics is needed to identify potential issues associated with their viability/desirability for potential implementation in commercial reactors.

Project Description:
The analysis effort provides support to identify promising advanced fuels/ATF candidates for further development. The project performs analytical assessment that includes neutronics analyses to evaluate the impact on performance parameters (e.g., cycle length/burnup, power distributions, etc.), safety-related characteristics (e.g., reactivity and
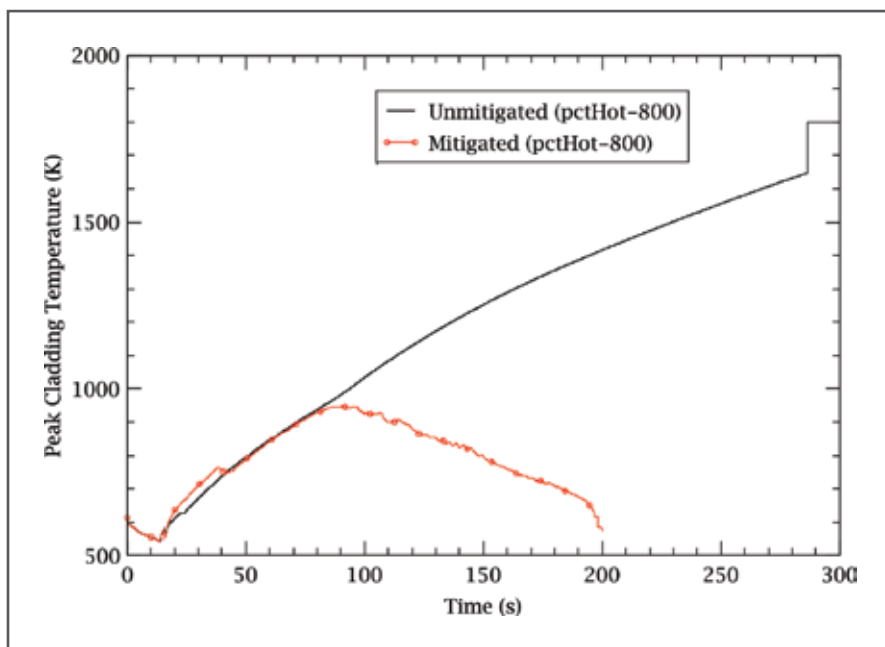
Figure 2. Peak Cladding Temperature (FeCrAl) in a Large-Break Loss-of-Coolant Accident (LB-LOCA)

control coefficients/worths, kinetics parameters, etc.), and performance in a broad spectrum of transient and accident scenarios. The key components of the assessment are initial screening analyses, three-dimensional core analyses, and reactor system analyses. The integrated analytical approach starts with an initial screening using infinite lattice physics calculations. Three-dimensional core analyses then provide fuel cycle performance data and reactor kinetics parameters for time-dependent accident analysis.

Reactor system analyses finally provide estimates of safety margins and "coping time", i.e., time to various limiting/ failure conditions, under various transient accident scenarios. The search for attractive and viable advanced fuels/ ATF candidates requires consideration of a broad spectrum of potential fuel and cladding options. Evaluation of the performance of these fuel and cladding options is vitally important, because it identifies whether the options have at least equivalent performance to the present UO_2-Zr fuel-cladding

system under nominal conditions and improved performance in accident scenarios. Both of these characteristics are necessary in order for any ATF concept to be adopted by industry and utilities.

Accomplishments:

The main activities in FY18 included: 1) a White Paper assessing the ability to calculate "coping time", and 2) continued assessment of advanced fuel concepts for LWRs.

The White Paper "Assessment of Ability to Determine Time Available to Recover from Accidents for ATF LWR Fuel Concepts (i.e., "Coping Time")" was prepared including input from Oak Ridge National Laboratory (ORNL), Massachusetts Institute of Technology (MIT), Idaho National Laboratory (INL), and Electric Power Research Institute (EPRI). Because accident scenarios

differ between reactors, and accident progression can take different paths, several definitions/figures-of-merit for coping time and their bases are described, followed by a brief description of potential accident scenarios to be considered. The phenomena that need to be modelled, and the computational tools available to perform the analyses are described. Tools utilized by industry are identified, but the assessment of capabilities is focused on tools utilized by the national laboratories; namely, TRACE, TRACE-BISON, and MELCOR. The data needed to "accurately" model ATF concepts in the simulation of the initial conditions (i.e., the condition of the reactor at the initiation of the accident) and the evolution of the accident are also described.

Several options for coating fuel pellets to mitigate the fuel/coolant interactions for water-reactive high-density

This project evaluates the performance of advanced fuel/ATF options for LWRs, and identifies whether the options have at least equivalent performance and safety characteristics relative to current uranium dioxide (UO_2) fuel and zirconium alloy (Zr) cladding, as well as offering potential performance and/or safety benefits under nominal conditions and in accident scenarios.

fuel phases, such as U_3Si_2, and UN have been proposed by Los Alamos National Laboratory (LANL). Initial modelling of these options with the TRITON neutronic lattice physics code to provide an initial estimate of the impacts on reactor performance and safety characteristics are underway. Figure 1 shows K-infinity vs. burnup for 20 μm coatings of Zr-metal, ZrB_2, Y_2SiO_5, Cr-10Al, and AlN on UN. The analyses all assumed the standard Westinghouse 17x17 assembly geometry and fuel pellet OR and Zircaloy cladding IR/OR; therefore, the gap was reduced by the thickness of the coating. The nitrogen for all cases was "natural" as was the boron in the Zirc-diboride. The burnups for the different coatings are all essentially the same, and only slightly lower than for the "reference" UO_2-Zr case. The reactivity and control coefficients (fuel and moderator temperature coefficients, and soluble boron and control rod worths) are effectively the same for all the coatings for the 20 μm thickness. Analyses for a 40 μm thickness coating and for U_3Si_2 fuel are underway.

Transient analyses of ATF concepts with the TRACE systems analysis code continued. As a result of a benchmark effort the TRACE 3-loop PWR plant model was updated with emergency core cooling systems (ECCS) injection characteristics that are prototypic of 3-loop plants. Using this updated model a large-break loss-of-coolant accident (LB-LOCA) was analyzed. The analyses used a version of TRACE version 5 Patch 5 modified by MIT to include FeCrAl and several other metals as build-in materials. The MIT modification includes metal/water reaction models for these new build-in materials. Two cases were analyzed for a core consisted of UO_2 fuel rods cladded in FeCrAl, a mitigated case and an unmitigated case. The mitigated case assumed availability of the full complement of ECCS while the unmitigated case assumed no ECCS activation. Results for the peak cladding temperature and the maximum equivalent clad reacted (ECR) are shown in Figures 2 and 3 for the peak cladding temperature (PCT), and ECR, respectively. For the hot rod in the unmitigated case the FeCrAl cladding reached the melting temperature of 1800 K at 286 second and the corresponding ECR at incipient melting was 8.6%. The mitigated case did not result in significant clad heatup and there was no clad reaction.

Analysis of SiC BWR Channel Box Performance under Normal Operating Conditions

Principal Investigator: Gyan Singh, Oak Ridge National Lab; University of Tennessee, Knoxville
Collaborators: J. Gorton[3], Danny Schappel[2], Nick Brown[3], Yutai Katoh[1], Brian Wirth[1,2], and Kurt Terrani[1];
[1] Oak Ridge National Lab, [2] University of Tennessee, Knoxville, [3] Pennsylvania State University

Figure 1. a) Spatial temperature distribution of the channel box in the simulation. Walls 2 and 3, and 1 and 4 share borders. b) Spatial distribution of fast flux in the channel box for the simulation.
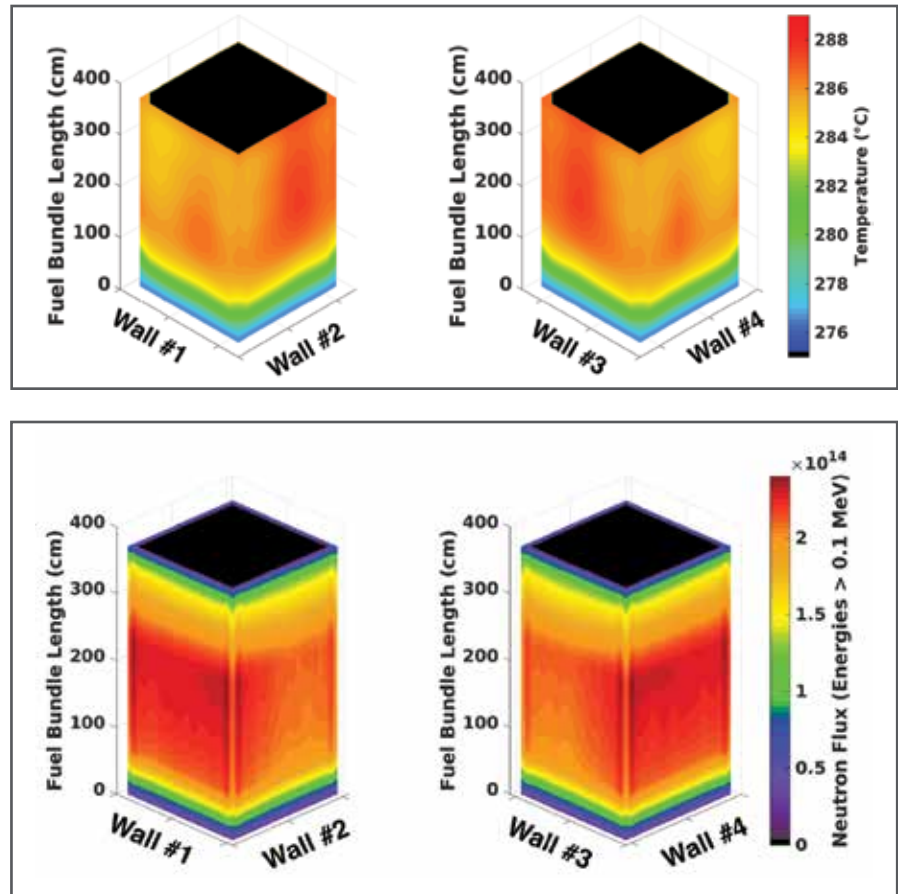
Replacement of the reference zirconium channel box with a silicon carbide composite (SiC-SiC composite) provides the potential to significantly reduce hydrogen generation in a boiling water reactor (BWR). Qualification of new channel box materials encounters many challenges similar to those of a new cladding material. Dimensional stability is a paramount requirement of channel box materials given the impact to thermohydraulic performance and core cooling. The swelling behavior of SiC–SiC composites is therefore a critical factor in deployment. Analysis

We unveiled (experimentally and theoretically) details of grain boundary scattering and its impact on the thermal conductivity in uranium dioxide.

of these responses is complicated by the significant temperature and flux gradients that are developed over the length of the channel box. This project aims to demonstrate and refine coupled modeling approaches to understand channel box deformation to guide execution of future irradiation testing.

Project Description:

A multi physics thermal-mechanical analysis was performed to evaluate the performance of silicon carbide composite fuel channel box in a BWR. The analysis involved providing neutronic and thermal-hydraulic boundary conditions using the Serpent and CTF codes, respectively, to inform a finite element thermal-mechanical analysis using ABAQUS and BISON. Figure 1 presents the temperature and fast (E>0.1MeV) neutron flux profiles in the BWR channel box. The neutronic-thermohydraulic model contained 119 coolant sub-channels, 92 fuel rods, and space for two large water rods. Seven axial locations using 2D geometry were

selected for the neutronic-thermohydraulic calculations, and the results at these locations were interpolated to form a 3D map of the temperature and neutron flux.

Accomplishments:

A channel box was created in the models by placing a heat slab structure with thermal properties of nuclear-grade SiC-SiC composites around the perimeter of the fuel pin lattice. The geometric dimensions for the model were representative of a GE14 BWR assembly. The outer cross-sectional width was 14.02 cm, the wall thickness was 3.05 mm, and the height of the channel box was 371 cm. Default material properties provided in CTF for UO2 fuel and zirconium-based cladding were implemented. The spacer grids are modeled based on BFBT 8x8 spacer grids with modified loss coefficients. The average axial power profiles for the model were obtained from the literature, and the radial heterogeneity in the power and neutron flux is directly included from

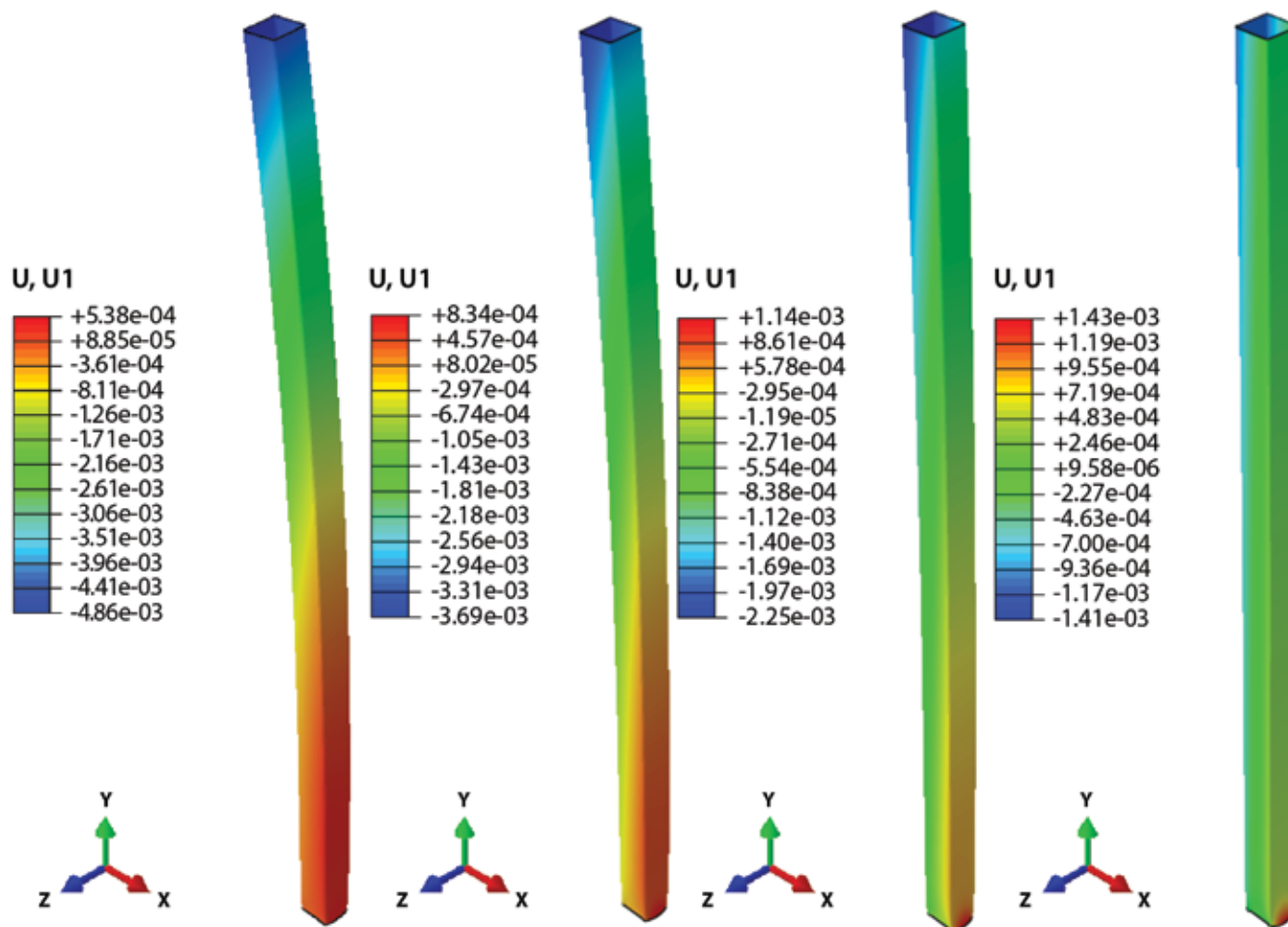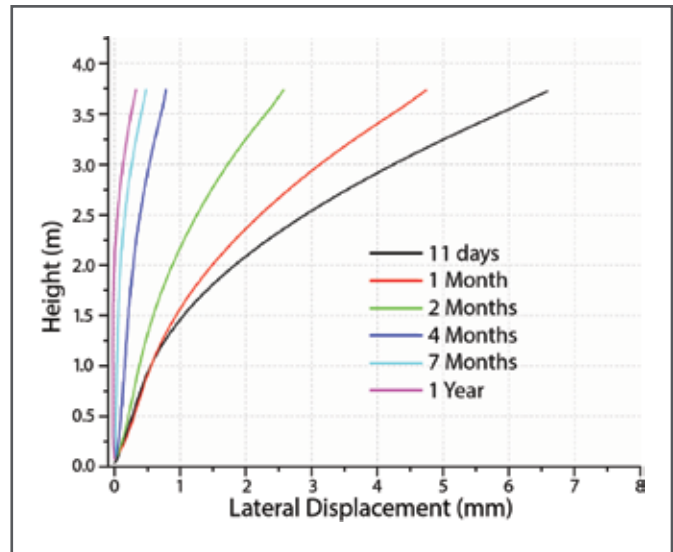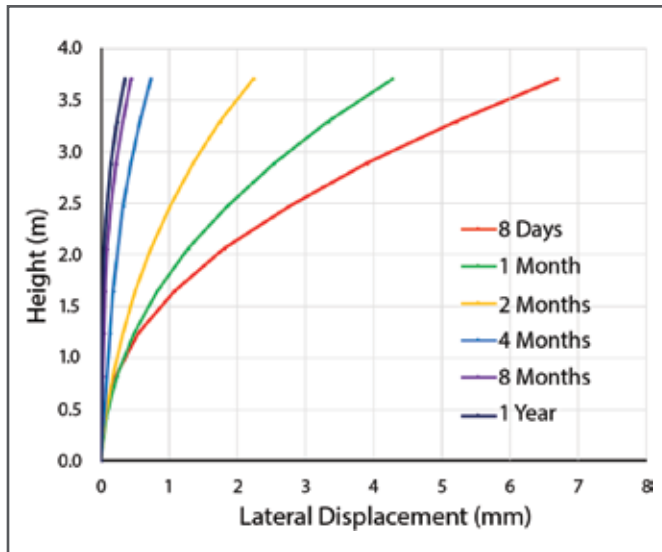| U, U1		U, U1		U, U1		U, U1	
+5.38e-04		+8.34e-04		+1.14e-03		+1.43e-03	
+8.85e-05		+4.57e-04		+8.61e-04		+1.19e-03	
-3.61e-04		+8.02e-05		+5.78e-04		+9.55e-04	
-8.11e-04		-2.97e-04		-2.95e-04		+7.19e-04	
-1.26e-03		-6.74e-04		-1.19e-05		+4.83e-04	
-1.71e-03		-1.05e-03		-2.71e-04		+2.46e-04	
-2.16e-03		-1.43e-03		-5.54e-04		+9.58e-06	
-2.61e-03		-1.81e-03		-8.38e-04		-2.27e-04	
-3.06e-03		-2.18e-03		-1.12e-03		-4.63e-04	
-3.51e-03		-2.56e-03		-1.40e-03		-7.00e-04	
-3.96e-03		-2.94e-03		-1.69e-03		-9.36e-04	
-4.41e-03		-3.31e-03		-1.97e-03		-1.17e-03	
-4.86e-03		-3.69e-03		-2.25e-03		-1.41e-03	

Figure 2. Displacement (meters) of the channel box along the x-direction (perpendicular to wall) after 11 days, one month, two months and four months (from left to right). Displacement is exaggerated for visualization purpose. The total displacement is shown in Figure 3.

detailed lattice physics calculations. A power profile for each rod was generated to account for the radial power variation in the assembly.

To ensure the accuracy of the models, comparisons were made to available literature data on two-phase flow and boiling in BWRs. The Serpent and CTF models were compared with void fraction results as a function of flow quality provided literature data, and good agreement was obtained between the

modeling predications and the experimental results. Additionally, the Serpent and CTF coolant density predictions were compared with predictions from TRACE/PARCS provided by the literature and reasonable agreement was obtained.

As shown in Figure 1, a modest temperature variation exists around the perimeter of the channel box, with a maximum temperature on walls 2 and 3, while the minimum temperature exists diagonally across the channel

Figure 3. Plots comparing the lateral displacement magnitude predicted by BISON (left) and Abaqus (right).

box on walls 1 and 4. Notably, the fast neutron flux has a different behavior in which the maximum neutron flux occurs on walls 1 and 4. Finite element modeling of the channel box deformation due to the thermal-mechanical conditions was performed using both ABAQUS and BISON, and the results are shown in Figures 2 and 3.

The channel box is predicted to undergo several millimeters of bowing due to the transient neutron irradiation-induced swelling of the SiC-SiC composite. This transient swelling behavior consists of a rapid increase to about ~2% volume increase at relevant BWR coolant temperatures and subsequently the magnitude of the swelling saturates, but with a neutron flux and temperature dependent saturation dose. The magnitude of the swelling is inversely propor-

tional to temperature, however in this situation the temperature gradients did not dominate the results. The regions with higher neutron fluxes swelled sooner than those with lower levels, and this resulted in the lateral displacement. Since the circumferential temperature variation was fairly small, the transient swelling was predicted to saturate at very similar levels, which resulted in a relaxation of the lateral displacement. Thus, the maximum lateral displacement was predicted to be about 6.5 mm and occur at 8 to 11 days. Then over the course of the next year, the lateral displacement was predicted to reduce to about 0.3 mm.

Elucidating cladding behavior under unmitigated large-break loss-of-coolant accident conditions using the BISON fuel performance code

Collaborators: Ryan Sweet, Andy Nelson, Kurt Terrani, and Brian Wirth

> *Coupled BISON & TRACE modeling efforts improve understanding of cladding behavior during large break loss of coolant accidents and informs design of transient testing.*

The main goal of identifying an alternative cladding material to Zircaloy for Light Water Reactor (LWR) fuel systems is to improve the reactor safety during high-temperature transient conditions. In order to provide insight into how FeCrAl cladding will perform compared to Zircaloy in this environment, fuel performance code capabilities can be extended, provided that thorough constitutive models are developed and implemented. The target of this analysis is the cladding behavior in the high-temperature environment sustained during a potential large-break loss-of-coolant (LBLOCA) accident. In this analysis, conditions from a mitigated LBLOCA are extended to compare the beyond design basis accident response of FeCrAl as compared to standard Zircaloy cladding.

During a loss-of-coolant accident with Zircaloy cladding, as the reactor loses its capability to cool the fuel, fuel rod temperatures begin to increase. Eventually, if this increase in temperatures is unmitigated, then the cladding will begin to 'balloon', or deform outwards, due to the pressure differential between the interior of the fuel rod and the pressure remaining in the reactor pressure vessel. If this deformation persists, cladding burst can occur during these high-temperature conditions from a combination of thermal creep and plasticity. If the temperature continues to increase further, the exothermic oxidation reaction of the Zircaloy cladding may become autocatalytic. This produces large amounts of heat, consumes the cladding, and produces hydrogen gas. Within this study, the cladding performance for both FeCrAl and Zircaloy cladding under these conditions are targeted.

Project Description:
Traditionally, separate fuel performance codes are utilized to simulate the fuel rod behavior under steady-state and transient conditions either in conjunction with each other, or to provide a stand-alone analysis. Transitioning the state of the fuel rod from normal operation into the transient environment is necessary to evaluate the unique condition of the fuel at a variety of fuel burnups. To accomplish this, these simulation conditions contain long-term steady-state reactor operation before transitioning into the accident scenario. This allows the state of the integral fuel rod at different specified burnups to be incorporated into the transient analysis.

To accurately simulate the cladding behavior under a loss-of-coolant accident, several key phenomena must be incorporated, including: chemical and phase changes of the cladding alloy (such as oxidation), high-temperature constitutive behavior,
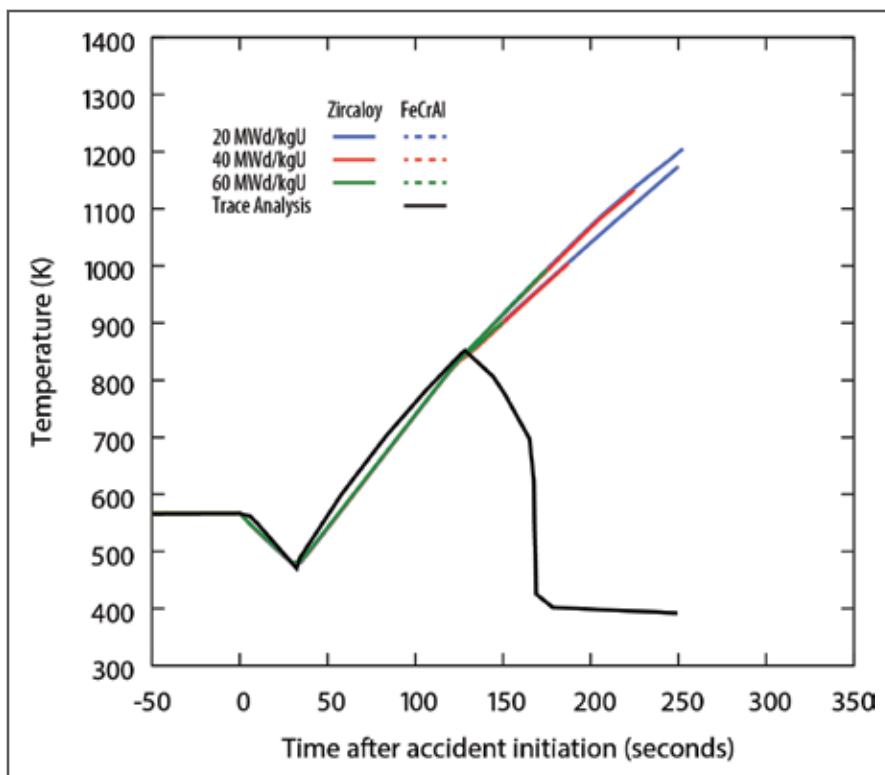
Figure 1. The maximum cladding temperature for both cladding types as a function of time after the accident initiation. The results from the TRACE simulation are (black line) included in to provide a comparison.

and rupture criteria for both cladding materials. These simulations use previously-implemented properties for the Zircaloy cladding, and newly-implemented properties for the FeCrAl cladding, as available. Data used to develop the high-temperature plastic behavior and rupture models for FeCrAl are derived from experiments performed at Oak Ridge National Laboratory (ORNL).

Accomplishments:

The fuel rod geometries used in these simulations are representative of a Boiling Water Reactor (BWR)/4. Because of the parasitic thermal neutron absorption from the FeCrAl alloy cladding, the cladding is thinned while the fuel radius is enlarged to partially offset the neutronic penalty. The 2D-axisymmetric simulation

Figure 2. The maximum cladding hoop stress is reduced as the accident begins, however, the hoop stress in the FeCrAl cladding monotonically increases as the accident progresses.
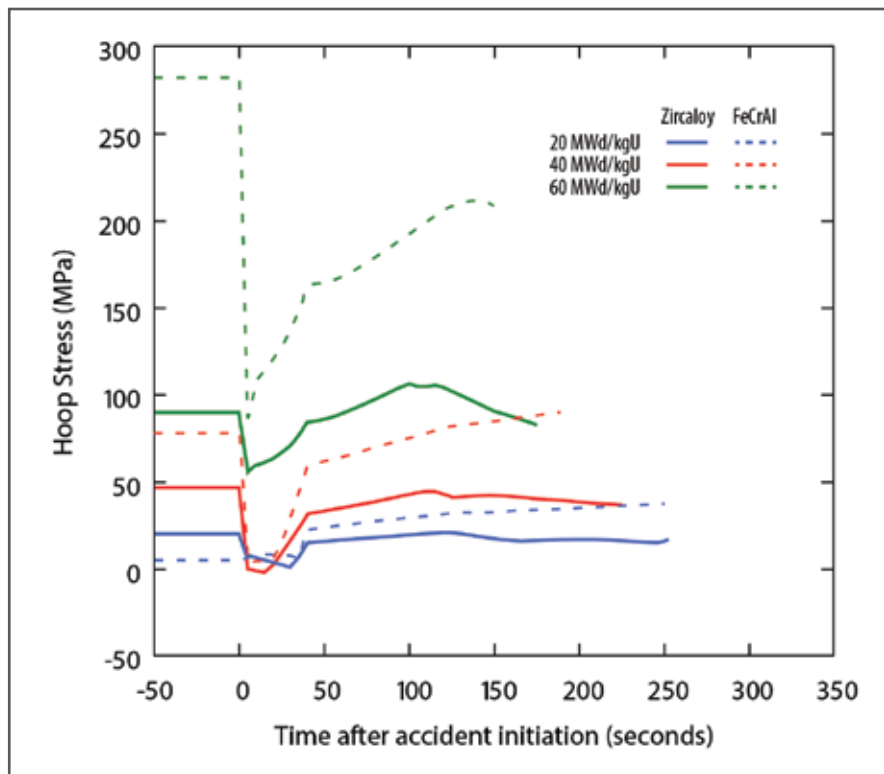
capability in BISON is utilized, and, as such, this analysis considers azimuthally uniform heat transfer conditions.

Before the onset of the transient conditions, the fuel rod is operated under steady-state conditions. When the fuel reaches a specified fuel burnup, the accident conditions are initiated. These conditions are modified from a previous study which used the TRACE thermal-hydraulics analysis tool, and are designed to simulate a reactor scram and subsequent loss of coolant inventory and flow. The cladding and fuel temperatures begin to increase as the decay heat from the fuel can no longer be removed from the reactor vessel.

Figure 1 shows the maximum cladding temperature during the transient. In this figure, the reactor is scrammed at 0s, and the reactor power then decays according to the fission product decay heat relation. The coolant flow slowly decreases as the core recirculation pumps stop. As

the coolant flow stagnates, the coolant temperatures increase, and the coolant heat transfer coefficient is decreased.

The cladding temperatures slowly decrease after the reactor has scrammed, reaching a minimum at ~41 seconds. This plot also shows the cladding temperatures from the TRACE simulation that actuated the low-pressure injection at approximately 91 seconds. In the TRACE simulation the cladding temperatures begin to decrease at ~140 seconds as the core is re-flooded. However, in this BISON analysis, the temperatures continue to increase until the cladding fails. These simulations are terminated when the burst criterion is reached. These results show progressively lower temperatures for the increasing fuel burnups and very similar burst times for both cladding types.

Figure 2 shows the maximum cladding hoop stress during the accident conditions simulated with BISON. Leading up to the accident scenario, the maximum cladding hoop stress is initially in a tensile state due to mechanical interaction between the fuel and cladding. After the onset of the LBLOCA conditions, the gap is reopened, as the reactor coolant

pressure is reduced to atmospheric pressure. This generates significant hoop stresses in the cladding, as the fuel rod plenum pressure begins to increase with the fuel temperatures. In these simulations, the FeCrAl cladded fuel rods fail at much larger stresses their Zircaloy counterparts.

These results show that the FeCrAl cladding will generally burst at a similar time and temperature as the Zircaloy cladding under the specific reactor operating and accident conditions that were simulated, but experience significantly larger cladding hoop stresses. As expected, the main driving force for differences in burst behavior between the various fuel burnups is the pressure differential across the cladding as the transient conditions evolve. The FeCrAl cladding thickness used in this analysis is thinner than the thickness expected to be deployed in commercial reactors, which would tend to increase the difference in cladding failure time for the FeCrAl rods during the accident. To further enhance this analysis, additional improvements to the FeCrAl constitutive models and operating conditions are currently underway.

2.4 ATF CLADDING AND COATINGS

Production of additively manufactured components for advanced reactor development

Principle Investigator: Dr. Niyanth Sridharan (ORNL)
Collaborators: Dr. Kevin G. Field (ORNL)

High chromium ferritic-martensitic (FM) steels are candidate materials for a range of advanced nuclear reactor concepts due to their excellent swelling resistance even to high damage doses (>50 displacements per atom – dpa). FM steels have been manufactured for nuclear applications using traditional processing routes for decades, but issues still exist including the formation of intercritically-heated zones leading to deterioration of material properties. In addition, traditional processing routes limit the geometries possible and thus complex component geometries which could improve reactor efficiency and safety are not possible. Recent advances in additive manufacturing (AM) processes, including laser-powder blown directed energy deposition (DED), could provide a new paradigm for FM steels production and use within the nuclear industry. To investigate AM as a novel processing routes for FM steel, laser-powder blown DED manufactured structures from alloy HT9 was completed and then systematically tested and characterized.

Project Description:
Alloy HT9 – a 12Cr-1MoVW FM steel – is under consideration for advanced nuclear reactor concepts due its low swelling rates under irradiation as demonstrated under multiple programs including those within the Fast Flux Test Facility (FFTF). The swelling resistance and radiation tolerance of HT9 is primarily attributed to the complex microstructure in the wrought condition which includes martensite laths, high dislocation density, and a mixture of different precipitate phases including $M_{23}C_6$ carbides and MX precipitates. Production of this microstructure is typically controlled using a series of heat treatments to both normalize the alloy and then temper the martensitic structure.

Recently, the nuclear industry has shown interest in applying AM processes to produce components which replicate the wrought-like microstructure of varying FM steels including HT9. AM is a layer-wise building of a component that is completed by depositing metal powder using fusion methods. One process, laser-powder blown
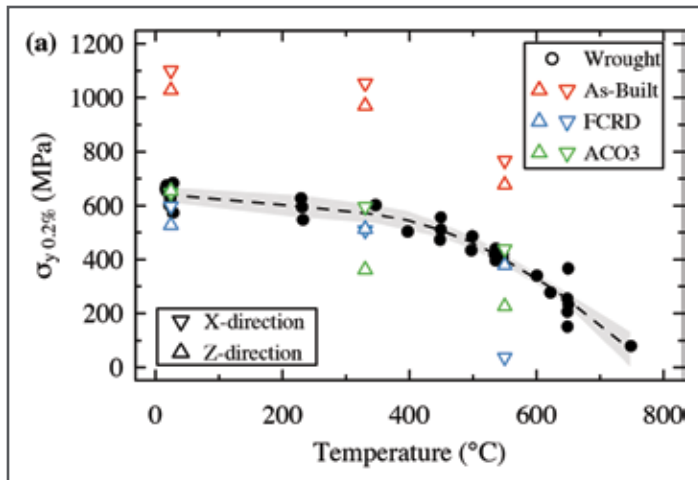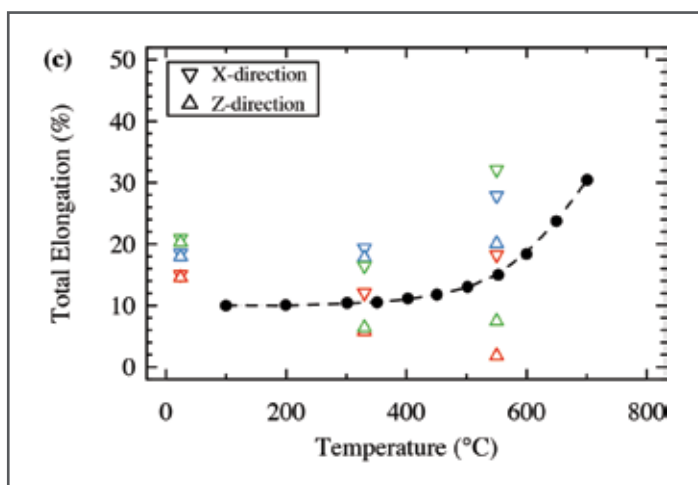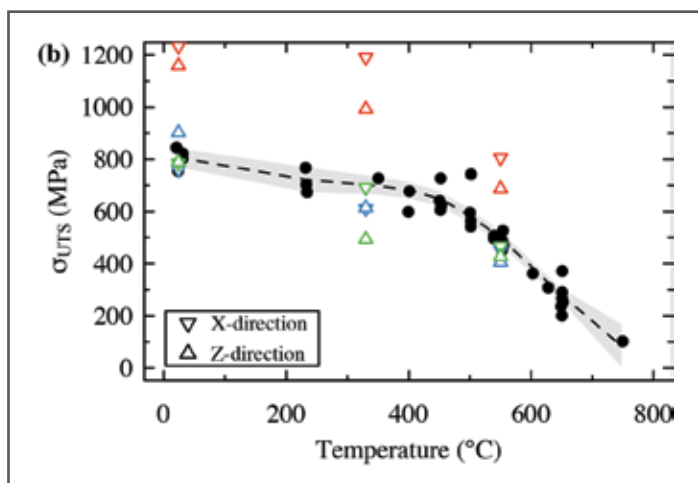
Figure 1: Comparison of historical wrought HT-9 to additive manufactured HT-9 in the as-built condition, FCRD heat treatment, and ACO3 heat treatment tensile properties including (a) yield strength - $\sigma_{y\,0.2\%}$, (b) ultimate tensile strength - σ_{UTS}, and (c) total elongation.

DED creates these layer-by-layer components by injecting micron size powder through a nozzle into a melt pool that is created by a high-powered coaxial laser beam. The system is capable of moving in the X-, Y-, and Z-direction thus enabling a wide range of geometries to be produced. The layer-by-layer deposition of this AM technique results in a complex temperature-temporal profile for the manufactured component which can result in varying microstructural features on a range of micron and millimeter length scales. The result is a detailed study is required to understand how the laser powder blown DED process generates microstructures using alloy HT9 in prototypic geometries and how these microstructures control the material properties.

Accomplishments:

HT9 powder feedstock was used in the fabrication of test coupons. A section of the coupons were then reserved for testing and characterization while other portions underwent a post-fabrication heat treatment. Two heat treatments were investigated, the first being deemed the "FCRD" treatment which was austenitized at 1040°C for 30 minutes followed by air cooling and tempering at 760°C and the other being the "ACO3" heat treatment which was austenitized at 1065°C for 30 minutes followed by air cooling and tempering at 750°C. The as-built and the FCRD and ACO3 heat treated specimens where then tested for strength and ductility as shown in the attached

Figure. The samples generally showed good mechanical properties with the as-built structure demonstrating significantly higher strength than the traditionally processed wrought alloys within minimal effect on the ductility. Additional multi-scale characterization was completed on the specimens to determine the factors controlling the observed strength and ductility for the AM-produced test specimens. The characterization efforts showed that the increased strength in the as-built specimen can be attributed to the fine grain size and carbide distributions which were produced due to the unique heat-cycling occurring during the additive process. The heat-treated specimens showed microstructures more consistent with those seen in traditionally processed HT9 which explains the reasonable agreement in mechanical properties for the ACO3 and FCRD processed specimens in

the Figure in comparison to the traditionally produced wrought alloys. Based on the preliminary results, it is clear that defect-free FM steels can be effectively manufactured using laser-powder blown DED additive manufacturing with acceptable mechanical properties. Based on the preliminary task that has been performed, it is clear that powder blown additive manufacturing is a viable processing technique to fabricate large structures such as clads, wrappers and ducts for the next generation of advanced reactor concepts.

FeCrAl ODS Alloy Development for Fission Platforms

Principal Investigator: Sebastien Dryepondt
Collaborators: Caleb P. Massey, Maxim N. Gussev, Philip Edmonson, Kory D. Linton and Kurt A. Terrani
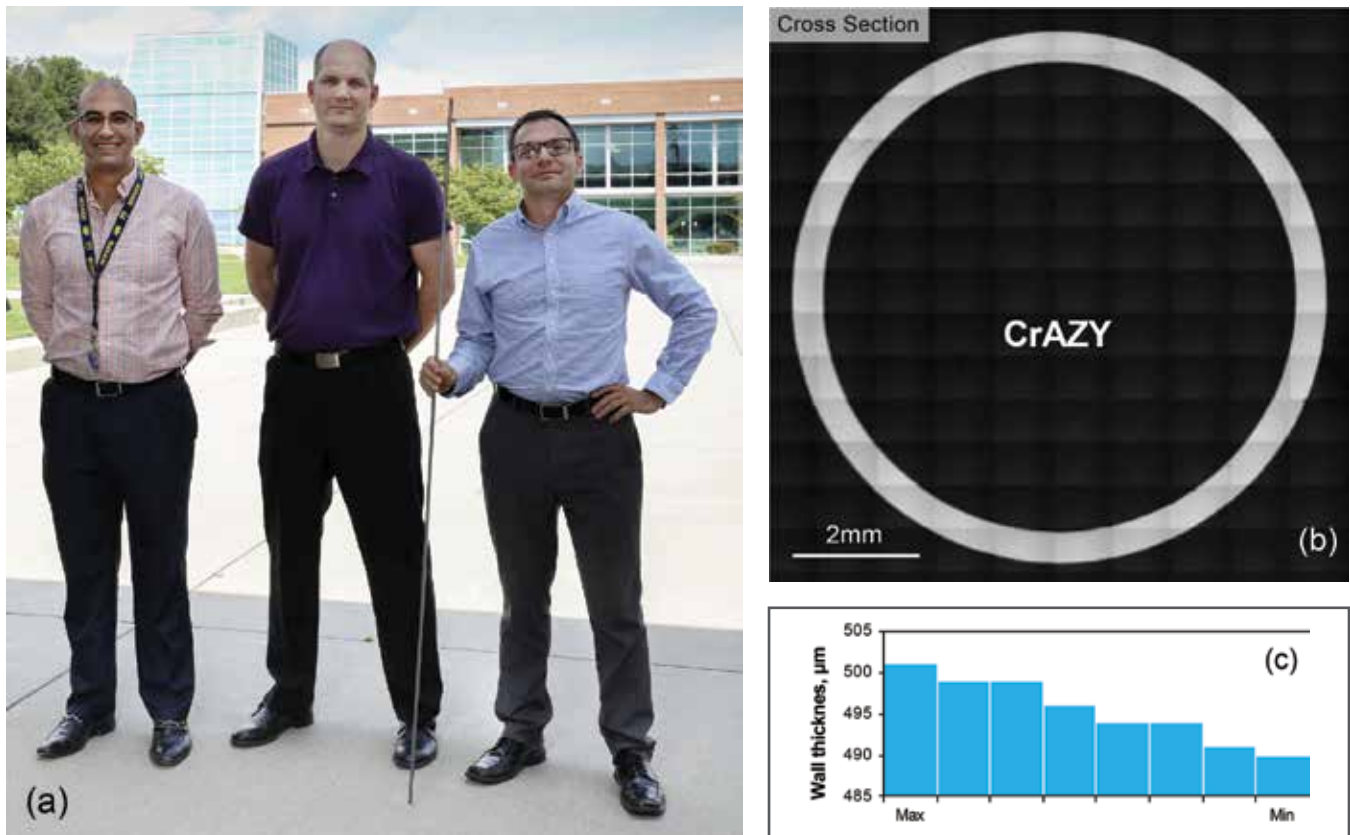

Figure1: a) CrAZY ODS FeCrAl tube, b) Cross-section of the tube, c) Wall thickness measurements showing the excellent dimension tolerance

Nuclear-grade FeCrAl alloys are leading candidates for accident tolerant fuel cladding and 200 feet of thin-walled seamless FeCrAlY tubes were sent to the Hatch nuclear power plant for in-service evaluation. The superior mechanical strength of oxide dispersion strengthened (ODS) FeCrAl alloys would allow for the use of thinner cladding to limit the neutronic penalty related to Fe-based alloys, and the high density of 2-4nm (Y,Al,O)-rich precipitates provide sinks for irradiation-induced point defects. The ODS FeCrAl high strength and great irradiation resistance result, however, in limited alloy ductility making thin tube fabrication challenging.

Project Description:

The project goal is to develop and fabricate new low-Cr ODS FeCrAl thin tubes for accident tolerant cladding application. The tubes need to exhibit great oxidation resistance in

steam at T>1400°C, good mechanical strength up to 1000°C, and superior irradiation resistance at 200-500°C. Optimization of the alloy composition based on the microstructure, oxidation behavior, tensile properties and irradiation resistance led to the selection of an ODS Fe-10/12Cr-6Al-0.3Zr+0.3Y2O3 alloy (named CrAZY). The project is currently focusing on the fabrication and characterization of 500um thick seamless ODS CrAZY tubes. ODS tube fabrication typically requires the fabrication of a master tube and then a succession of pilgering and annealing steps to decrease the tube thickness from several millimeters to 0.5mm. The process needs to be tailored based on the alloy microstructure and properties and may lead to anisotropic mechanical behavior for the final tube. Specific ring and tube specimens were designed at Oak Ridge National Laboratory (ORNL) to measure the tube room temperature tensile properties along the tube length and in the hoop direction.
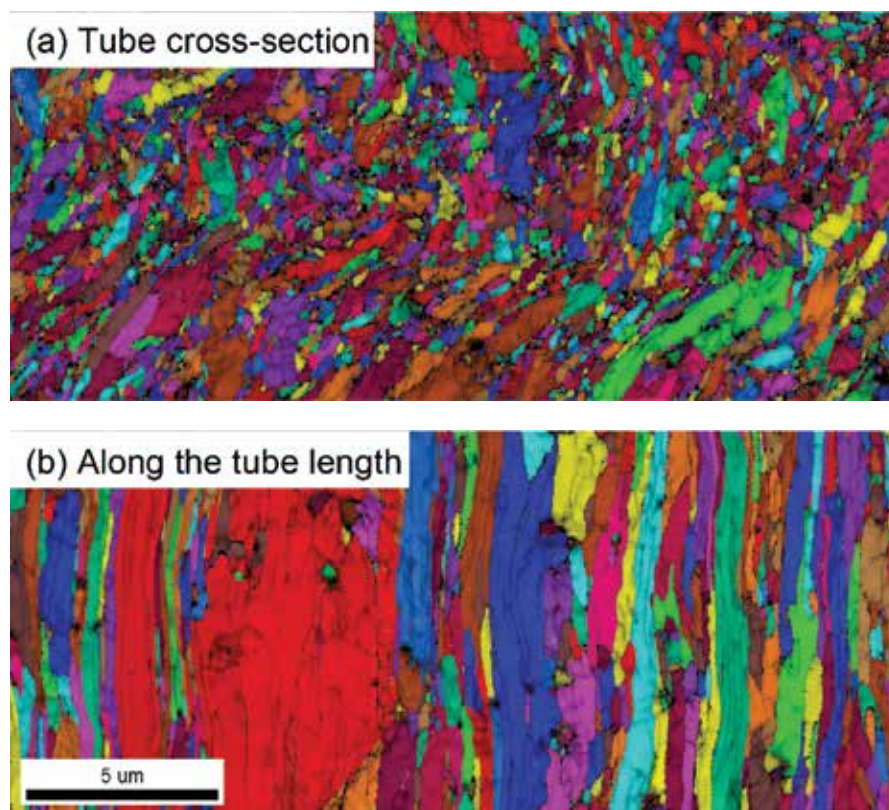

Figure 2: EBSD grain map of the CrAZY tube, a) Perpendicular to the tube length and b) along the tube length showing elongated grains and sub-grains along the pilgering direction
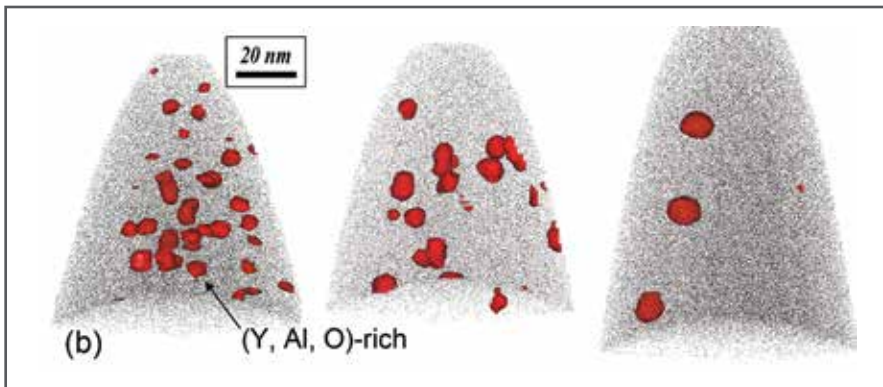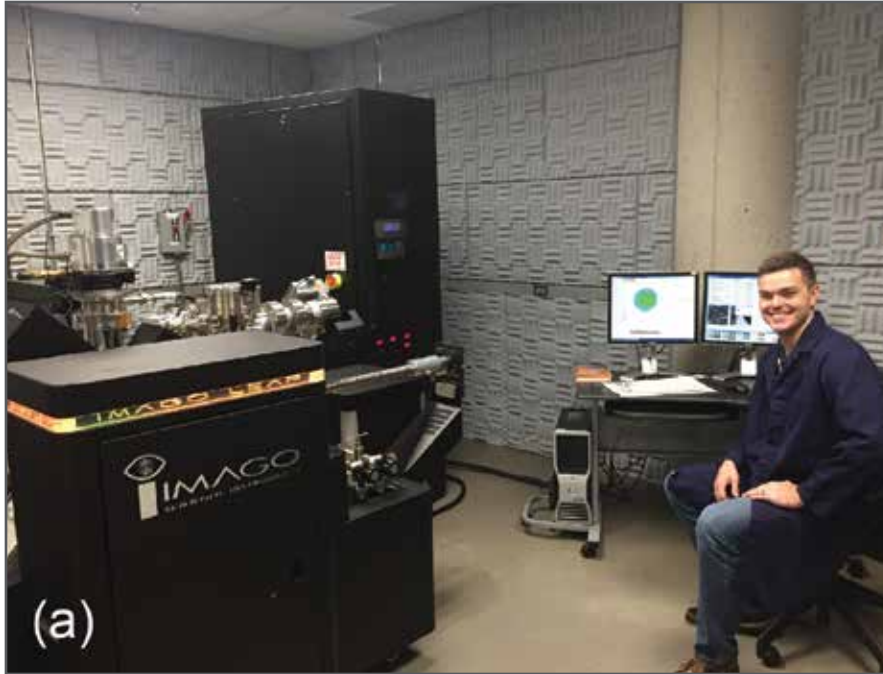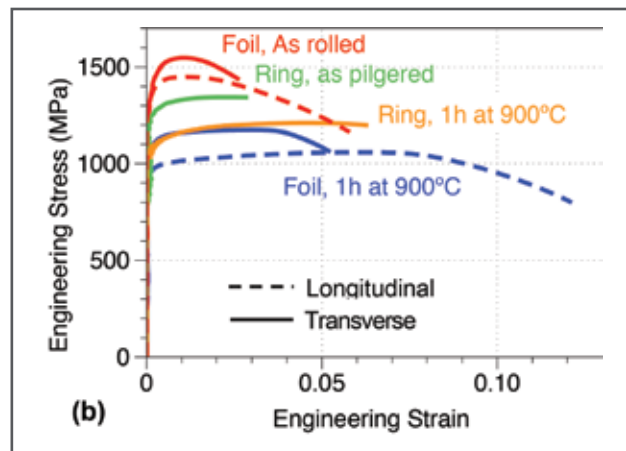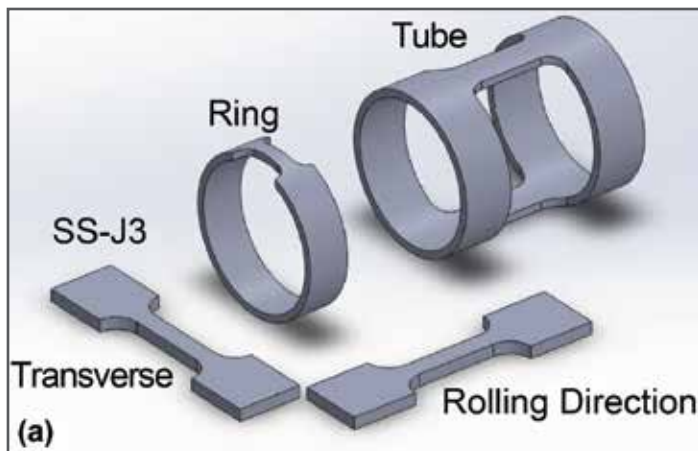
Figure3: a) Atom probe tomography (APT) LEAP 4000X, b) 3 APT maps showing local variations of nano precipitate density due to the 1100ºC extrusion temperature

The expected project outcome is the production of ODS FeCrAl cladding with great mechanical, oxidation and irradiation resistance providing enhanced safety margin compared to Zr-based and wrought FeCrAl cladding for the current and next generation reactors.

Accomplishments:

A new ODS FeCrAlYZr "CrAZY" tube, 1.68 m long, was fabricated by pilgering in collaboration with a Japanese tube manufacturer. After each pilgering step, microstructure characterization and hardness measurements were conducted on annealed pieces of tube to select the optimum heat treatments. The resulting tube showed excellent dimension tolerances, 500um +/- 4um for the wall thickness, and 8.47mm +/-0.01mm for the outside diameter. The extreme compressive stress during pilgering led to the formation of fine grains and sub-grains elongated along the pilgering direction, the latter being ~200-300nm in width and several micrometers in length., Atom probe tomography (APT) also revealed the presence of many (Y,Al,O)-rich precipitates in some areas, less than 5nm in size, but fewer precipitates were observed in other areas. This inhomogeneity is likely due to the high extrusion temperature required to decrease the master alloy hardness, 1100ºC, leading to the coarsening of some nano precipitates but may also be partly due to particle dissolution during the tube pilgering process.

Figure 4: a) Specimens used to characterize ODS CrAZY sheet and tube, b) Room temperature tensile curves showing the high strength but limited ductility of the ring specimen. Annealing for 1h at 900ºC improved the CrAZY tube ductility and decreased slightly the tube strength. Similar results were observed for thin CrAZY foils

The thin ODS FeCrAl "CrAZY" tube fabricated by pilgering exhibited very high strength at room temperature and great oxidation resistance in steam at 1400ºC

The tube microstructure resulted in very high longitudinal and hoop strengths at room temperature, but led also to anisotropic tensile properties with the ductility in the hoop direction being lower than along the tube. Post annealing treatment to reduce the tube residual stress was found to be a possible route to improve the tube ductility with a moderate reduction of the tube strength. Recrystallization of the tube with the formation of larger grains was observed for temperature as low as 900ºC, and the number of recrystallized grains increased with increasing the annealing temperature. All these results were consistent with previous experiments conducted on cold rolled foils. The CrAZY tube showed also good oxidation resistance in steam up to 1400ºC, and specimens have been machined to evaluate the irradiation resistance of the tube at the high flux isotope reactor (HFIR). On-going work aims at the fabrication of at least 5 new CrAZY tubes through a similar collaboration with an industrial partner. Slight modifications of the process parameters and alloy chemistry are expected to lead to further improvement of the tubes microstructure and properties.

Burst Testing of ATF Cladding Materials

Principal Investigator: Bruce A. Pint

> *Burst testing in steam of the new C26M FeCrAl alloy has shown superior burst behavior to first generation FeCrAl tubing and smaller burst openings at higher internal pressure.*

As part of the research and development process of replacing the current Zr-based fuel cladding for light water reactors with the more accident tolerant FeCrAl fuel cladding, one of the integral experiments needed to predict behavior is the large-break, loss of coolant accident (LOCA) burst test. FeCrAl has oxidation rates more than 100X slower than Zr up to near its melting point of near 1500°C. The burst test requires a ~30 cm long tube of the material with a uniform diameter and wall thickness. Thus, in the early stages of FeCrAl development, such tubing did not exist in sufficient quantity for this type of experiment. As the project matured and larger batches of the new FeCrAl tube were fabricated, it was now possible to conduct these experiments for the C26M alloy that was inserted into Plant Hatch in February 2018.

Project Description:

A key objective of the program is to develop reliable performance models for operation and during various accident scenarios for new accident tolerant fuel (ATF) concepts, such as the new FeCrAl cladding. Deploying this new oxidation-resistant ATF cladding will decrease the genera-tion of heat and hydrogen compared to Zr-based claddings and increase coping time in the event of an accident and allow more time for mitigation and evacuation. However, prior modeling of FeCrAl cladding used incomplete information on the physical properties of FeCrAl. This project is developing integral data and one incomplete area is the burst behavior of C26M, which is a 2nd generation FeCrAl alloy with improved strength due to the additions of Mo and Si. Tube burst is inevitable during large break LOCA conditions as tubes are internally pressurized for operation in the pressurized water operating. When the coolant (pressurized water) is lost, the tubes now have an internal stress not balanced by the outside environment. As the temperature increases and the material becomes weaker (and is consumed by oxidation due to oxidation in steam), the tube eventually bursts. The burst temperature and the size of the burst determine when and how much fuel might be released to the environment during various accident scenarios.

Accomplishments:

The goal in 2018 was to burst test C26M FeCrAl tube specimens from multiple batches and determine the performance of this new material relative to (1) first generation FeCrAl
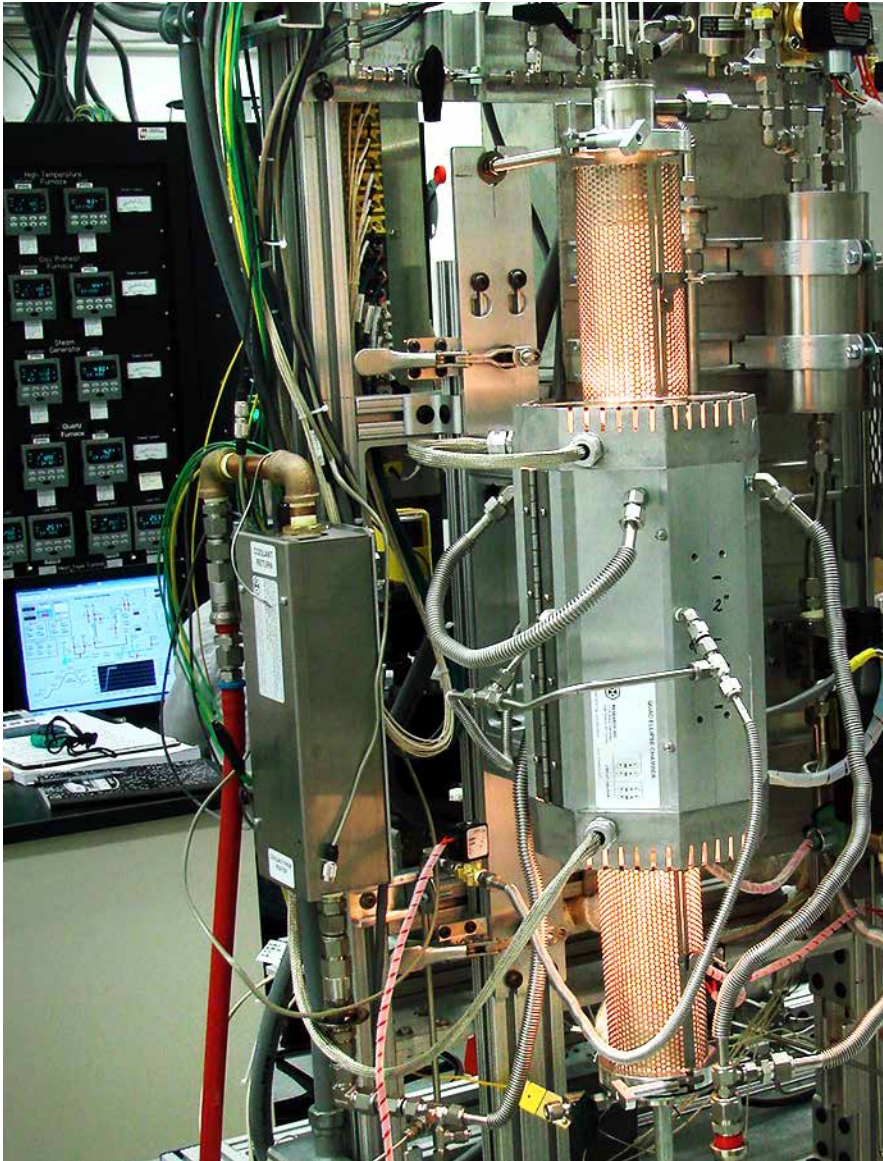
Figure 1: The integral loss of coolant accident (LOCA) system of the severe accident test station (SATS) at Oak Ridge National Laboratory

Figure 2: Burst temperature as a function of engineering hoop stress for various cladding materials examined in the SATS (data points from this year and prior years) alongside empirical correlations (lines) from the literature for Zr-alloys and type 304 stainless steel (SS).
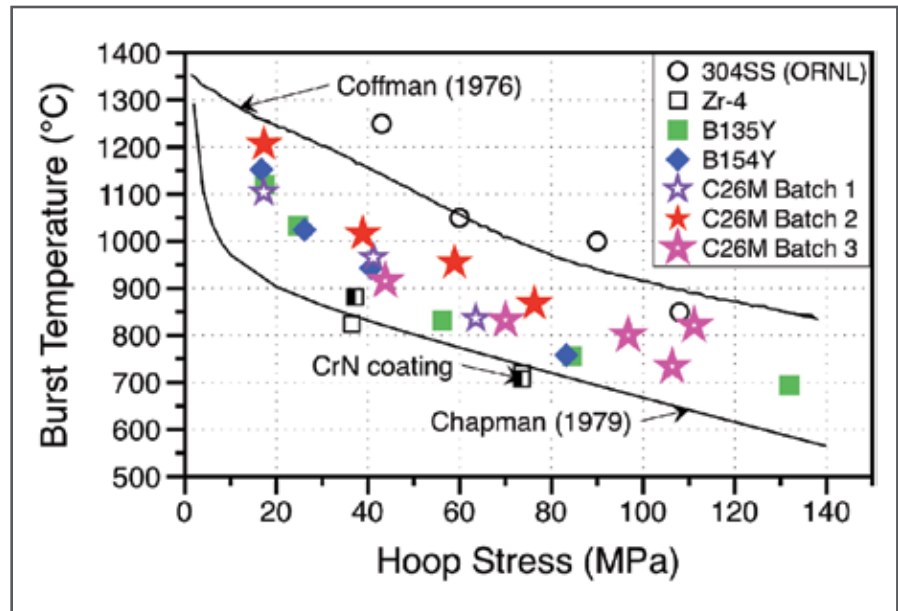
tubes (without Mo and Si additions) and (2) the decades of experience with stainless steel and Zr-based claddings. Figure (Pint)1 shows the integral LOCA furnace test equipment, which is part of the severe accident test station (SATS) at Oak Ridge National Laboratory (ORNL). The LOCA burst tests were conducted in the SATS on 30 cm long tube specimens that were filled with ZrO_2 pellets and internally pressurized to a fixed value followed by heating in steam to 1200°C at 5°C/s. The pressure was monitored in the tube during the experiment to determine the burst temperature. During cooling, water was injected into the system at 600°C. Figure (Pint)2 shows the prior work and the new results on three batches of C26M. The results are compared to results from the literature for Zr-based cladding and type 304 stainless steel. As the temperature increased in the experiment and the material got weaker, Zr-based alloys typically ballooned (i.e. expanded due to creep) and eventually burst. Previous characterization showed that the stronger FeCrAl alloys tended to just burst without significant deformation. The expectation was that the C26M tubing would outperform the "B" series tubing (e.g. designated B135Y (13Cr-5Al) and B154Y (15Cr-4Al) in Figure 2 because of the higher tensile properties attributed to the 2wt. %Mo and 0.2%Si additions. The

111 MPa   106 MPa   97 MPa   70 MPa
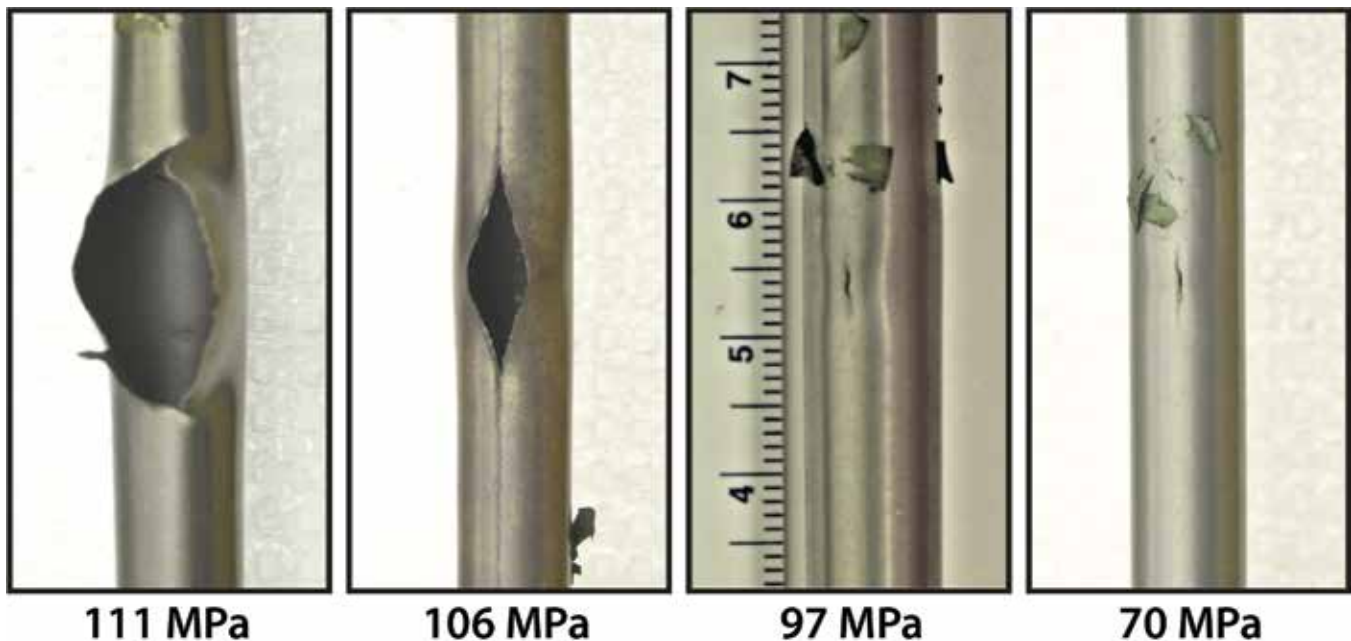
first batch of C26M specimens did not show improved burst temperatures compared to the 1st generation results and this was consistent with the low tensile properties reported [38], which were attributed to processing issues. In the 2nd batch of C26M tubing, the processing issues were resolved, and the burst temperatures were consistently higher than the 1st generation "B" alloy results. For the third batch of C26M tubing, several experiments were conducted at higher stress levels to examine the burst size. A few lower stress tests showed lower burst temperatures than expected. Those specimens are being characterized to determine if the tube wall thickness has been measured correctly, which could change the stress values in Figure 2. Figure 3 shows images of the burst tubes from four different stress levels shown in Figure 2. It is clear that above 97 MPa, the burst size increased dramatically. At 111 MPa, one of the ZrO_2 pellets that is a surrogate for the fuel pellets was ejected. For the "B" material, a large burst occurred at only 85 MPa so the C26M material is showing both an increase in the burst temperature and the stress where a large burst occurs. Additional testing is in progress to further study the burst behavior of C26M in the 90-130 MPa stress range and obtain statistical information on the burst behavior.

Figure 3: Images of C26M (FeCrAl) tubes burst at four different stress values. The small openings observed at lower stress values are marked with arrows. The locations where thermocouples (TC) were attached to the tubes is apparent.

Modified Burst Testing of ATF Cladding Materials

Principal Investigator: M. Nedim Cinbiz
Collaborators: N.R. Brown, R.R. Lowden Jr., M. Gussev, K. Linton, K.A. Terrani

Figure 1. (a) Schematics of the novel optical metrology technique, including a camera, a telecentric lens system, and first-surface mirrors. (b) Projections of SiC/SiC composite sample on the camera. The sample was dyed with black and white paint to create speckle patterns. (c) A schematic of the MBT for the PCMI-type loading.
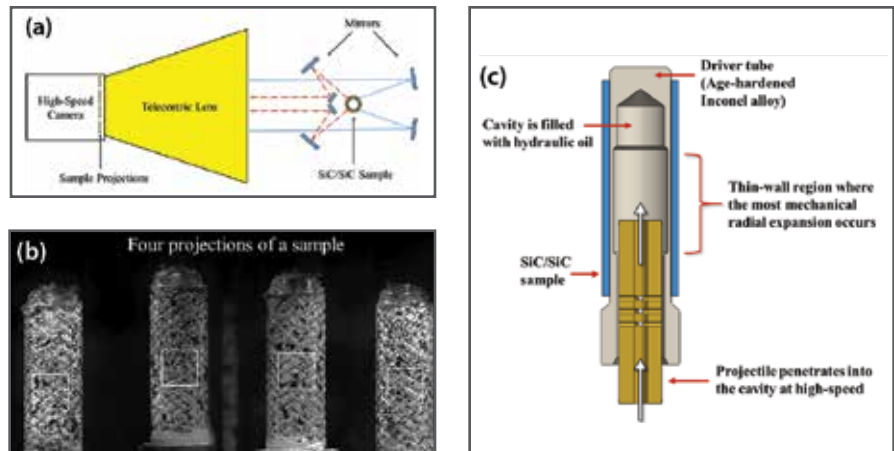
Accident tolerant fuel (ATF) cladding candidates must maintain or extend the safety envelope of current light-water reactors (LWRs) not only during severe accident conditions, but also during postulated design-basis accidents, such as a reactivity-initiated accident (RIA). During an RIA, the removal of the control rods from the LWR's core causes a prompt increase of the fission-rate density until the reactor is shut down. The deposited energy in the fuel increases the fuel temperature, which causes fuel thermal expansion. If the fuel and cladding are in contact before or during the RIA, this expansion imposes a mechanical deformation on the cladding's inner surface. This interaction, termed the pellet-cladding mechanical interaction (PCMI), occurs in milliseconds. During an RIA, cladding deformation may surpass the design limits of future ATF-installed LWRs. To address this issue, this study investigates the mechanical response of ATF cladding candidates under PCMI-like conditions over RIA time scales.

> *Modified burst tests of ATF cladding candidates with a novel optical metrology technique inform future in-pile transient testing of the cladding candidates' mechanical response during the pellet-cladding mechanical interaction phase of the RIA.*

Project Description:

The deformation and failure behavior of ATF cladding candidates during RIA relevant conditions must be addressed to ensure safe and reliable operation of ATF-installed LWRs. This study aims to support the design of the nuclear safety envelope for ATF cladding candidates in LWRs by performing out-of-pile transient tests. The out-of-pile transient tests investigate the mechanical behavior and failure characteristics of the ATF candidates under PCMI-like loading conditions at the strain-rates of postulated design-basis RIAs. This research benefits from the pulse-controlled modified burst test (MBT) instrument and a novel noncontact optical method for measuring mechanical strain. The MBT offers a uniform mechanical strain-driven loading that mimics PCMI, while pulse-control reproduces the pulse-widths of Transient Reactor Test Facility (TREAT) (from ~50 to >100 ms), CABRI (~20 ms), and Nuclear Safety Research Reactor (~10 ms) integral tests. The novel optical metrology technique enables 360° observation of the tubes, and tracks deformation or failure. ATF cladding candidates, including heat-treated (aged) iron-chromium-aluminum (FeCrAl) alloys and continuous silicon carbide fiber and silicon carbide matrix (SiC/SiC) composite tube samples have been tested to understand the failure characteristics of these materials during RIA relevant conditions.
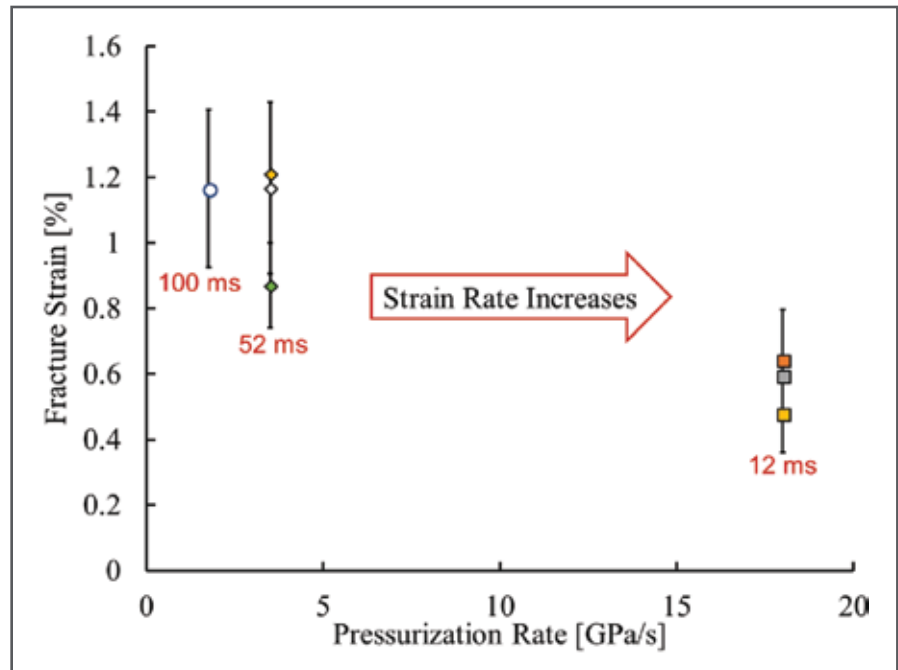
The results inform the design of the future in-pile experiments in the TREAT reactor. Thus, the out-of-pile transient testing capability is a vital tool for selecting ATF candidates to advance nuclear fuel safety.

Accomplishments:

The novel optical metrology technique was successfully applied to the pulse-controlled MBT as shown in Figure 1. The 360° view of the tube sample was tracked during the PCMI tests using a high-speed camera, a telecentric lens, and heat-resistant first-surface mirrors (Figure 1a-b). The working principle of MBT is depicted in Figure 1c: a projectile (core-pin) is actuated to move within the hydraulic oil-filled cavity of the driver tube. Motion of the core-pin causes pressurization of the hydraulic oil, triggering expansion of the driver tube, which imposed mechanical strain on the sample. This type of loading simulates the strain-driven nature of PCMI during postulated RIA.

The mechanical strains on the tube samples were calculated using custom DIC software. Validation of the DIC-strain measurement was performed by generating virtual images with speckle patterns at the prescribed strains and the loading paths. The DIC-calculated strains of 360°-wrapped pictures were compared with the prescribed strain values. The accuracy of the DIC-calculated strains was high ($\sim$150 μ) for the generated virtual images and the 360°-wrapped ones. The results indicated that the DIC-calculated strains from the wrapped images were in strongly significant agreement with the prescribed strain values.

ATF cladding candidates of SiC/SiC composite and long-term heat-treated (thermal aging) Gen-I FeCrAl alloy tubes were tested using MBT. SiC/SiC composite samples were tested at room temperature and FeCrAl alloys tubes were tested at 250 °C. The results of the SiC/SiC composite tests showed that the failure hoop strain decreased from 1.2% to 0.6% while the pressurization rate increased from 1.75 GPa/s to 18 GPa/s during PCMI-like loading (see Figure 2). This increasing pressurization rate corresponded to the decrease of the RIA pulse width from 100 ms to 12 ms. The thermally-aged Gen-I FeCrAl alloys failed in a ductile manner at rupture hoop strains of $\sim$6%. The ductility of the Gen-I FeCrAl alloy specimens was preserved during PCMI tests at RIA relevant strain-rates.

Multi-modal microscopy of irradiated materials for advanced reactor development

Principle Investigator: Dr. Kevin G. Field (ORNL)
Collaborators: Dr. Benjamin P. Eftink (LANL), Dr. Tarik Saleh (LANL), Dr. Stewart A. Maloy (LANL)
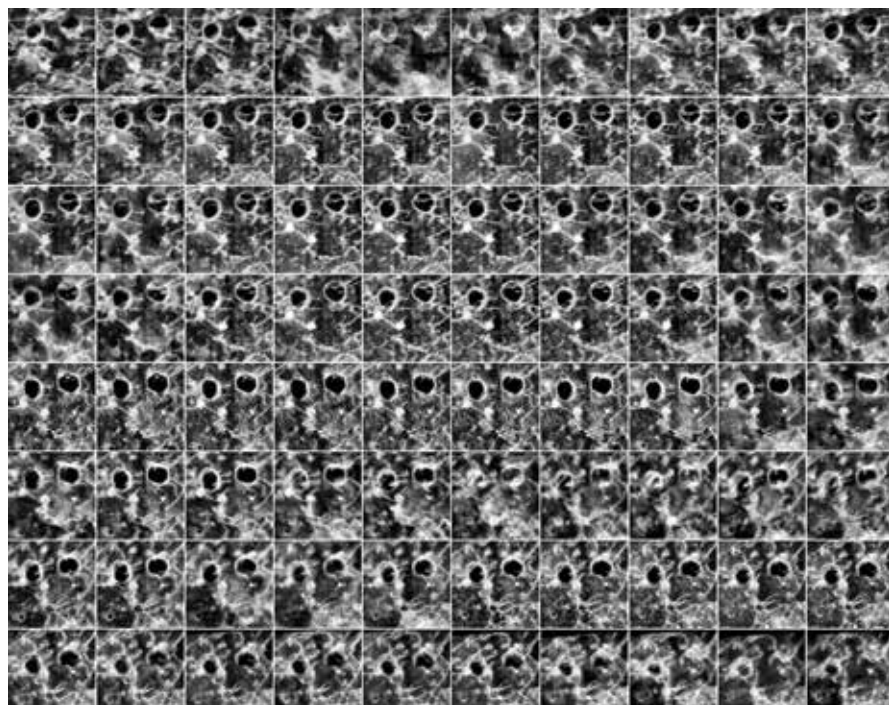
Figure 1. Image montage at varying specimen tilts taken at 1° increments from 40° to -40° in a left-to-right, top-down fashion. Individual image scale: 336×336 nm.

The deployment of advanced reactors will require materials which can withstand excess irradiation dose (>50 displacements per atom – dpa) at elevated temperatures for prolonged periods of time. Several candidate materials are under investigation for such applications including advanced ceramic-matrix composites and high chromium (>8 wt.% Cr) body centered cubic (BCC) ferritic/martensitic (FM) steels. FM steels are of significant interest due to their high resistance to radiation-induced swelling in typical advanced reactor conditions and generally low cost. To accelerate the development of the FM steel class, an alloy – HT9 – has been subjected to a series of advanced multi-modal microscopy experiments after high dose neutron irradiations to determine the root microstructural features which promote their high dose swelling resistance.

Project Description:

Neutron irradiation to FM steels generates a population of point defects (interstitials and/or vacancies) and small vacancy and interstitial clusters. Under elevated temperature irradiations these point defects and defect clusters can diffuse, annihilate, or agglomerate resulting in the formation of extended defects and promote additional phenomena such as radiation-induced precipitation. These defects and secondary features lead to degradation in the FM steels properties and could ultimately limit the lifetime of a component manufactured from FM steel in advanced reactor applications. The nucleation and growth of these radiation-induced or radiation-enhanced features is ultimately a complex process. To fully understand this process, and hence mitigate it, correlated microstructural examinations at length scales smaller than the width of a human hair are needed.

Significant strides have been made in the previous decade towards advanced microscopy techniques which can accurately detect the range of microstructural features and defects in FM steels including both atom probe tomography and electron microscopy. Furthermore more, the ability to conduct 3D characterization at ultra-fine (<10 nm) length scales using tomography-based techniques has rapidly progressed in the past decade. These advances have been systematically leveraged to further the current understanding of how FM steels perform under advanced reactor applications. Specifically, a heat of HT-9 (Heat 84425) which was neutron irradiated in the Fast Flux Test Facility (FFTF) to a damage dose of 155 dpa at 443°C was investigated using a state-of-the-art FEI F200X scanning/transmission electron microscopy (S/TEM) instrument at Oak Ridge National Laboratory (ORNL). 3D electron tomography was used to determine the spatial correlation between pre-existing microstructural features and features introduced under irradiation.

Accomplishments:

The FEI F200X S/TEM was used to generate two different 3D reconstructions with nanometer-scale resolution. The first being a reconstruction of diffraction-based STEM contrast and the second being of chemical signature-based contrast. The result is a full representation of the microstructure on a nanometer-scale voxel-generated image which included dislocation loops, line dislocations, G-phase, large and small cavities, and the Cr-rich □' phase. Excerpts from the diffraction-based STEM contrast reconstruction are shown in the attached Figure. The two large faceted-circles are voids within the structure while the bright-white contrast is a combination of both G-phase, dislocation loops, and line dislocations in the Figure. The 3D reconstructions allowed for the determination that the Cr-rich □' was denuded around the G-phase precipitates and the cavities within the region of interest leading to a "spotty" decoration of Cr-rich □' within the microstructure. The resulting conclusion was that Cr-rich □' does not co-locate with other microstructural features and does not promotes heterogeneous nucleation of void and cavity structures in FM steels irradiated to high dose at elevated temperatures. This provides new insight which will enable the development of more radiation-tolerant steels for advanced nuclear reactor concepts. In addition, the displayed advanced microscopy technique which relies on multi-modal characterization provides a new avenue for characterization studies within the Advanced Fuels Campaign (AFC).

Multi-modal electron microscopy-based tomography facilitates new insights into defect generation and stability in radiation-tolerant steels.

2.5 IRRADIATION TESTING AND PIE TECHNIQUES

Image Analysis Applied to Uranium Silicide Accident Tolerant Fuel Microstructure

Principal Investigator: Jason Harp
Collaborators: Fabiola Cappia

> *Integral PIE data from irradiation tests prototypical of LWR irradiation conditions are fundamental to confirm suitability of U_3Si_2 as ATF for advanced LWR.*

Image analysis is a powerful tool capable of quantifying microstructural features. It is extensively used in many fields, and in material science to relate physical properties at the micro-scale to the engineering ones at the macro-scale. When speaking about nuclear materials, the knowledge of their microstructure and how this changes under irradiation becomes a key safety factor, as the microstructure impacts the performance and safety under irradiation.

Image analysis is applied as postirradiation examination (PIE) technique to investigate the microstructure of U_3Si_2, an Accident Tolerant Fuel (ATF) candidate, irradiated at low burnup under Light Water Reactor (LWR) conditions in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL).

Project Description:

Accelerating the development of new fuels and cladding with enhanced safety and performance compared to the current fuel systems for present and future generations of Light Water Reactors (LWRs) is a priority of the Department of Energy, Office of Nuclear Energy (DOE-NE). A joint effort towards this goal has been undertaken by the DOE-NE, National Laboratories, Universities and industry within the ATF Campaign. Preliminary data regarding irradiation performance of the proposed new concepts are of paramount importance for qualification purposes, to down-select the most promising fuel systems and optimize the fuel development strategy. U_3Si_2 is one of the most attractive candidates, due to its high thermal conductivity and increased uranium loading compared to UO_2. The higher thermal conductivity reduces the temperature gradients and stresses within the fuel, implying that the fuel can be operated either with a higher margin to melting or at higher power that improves economic operation without compromising safety. The higher uranium loading allows reduction of initial enrichment, and allows the deployment of alternative cladding materials with higher neutron penalty compared to Zircaloy. Irradiation of U_3Si_2 rodlets has been carried out to provide first performance data of U_3Si_2 under commercial LWR conditions. In particular, assessment of fuel microstructure integrity and fission gas behavior are the highlight of this project, being those key fuel performance parameters of merit.
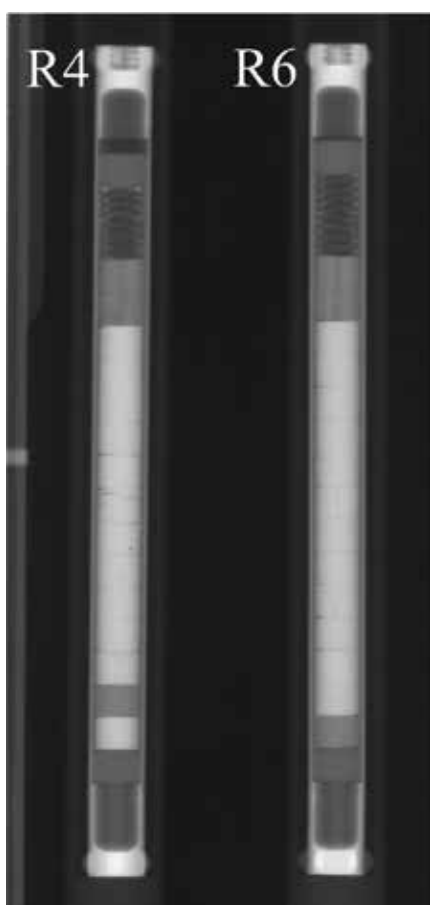
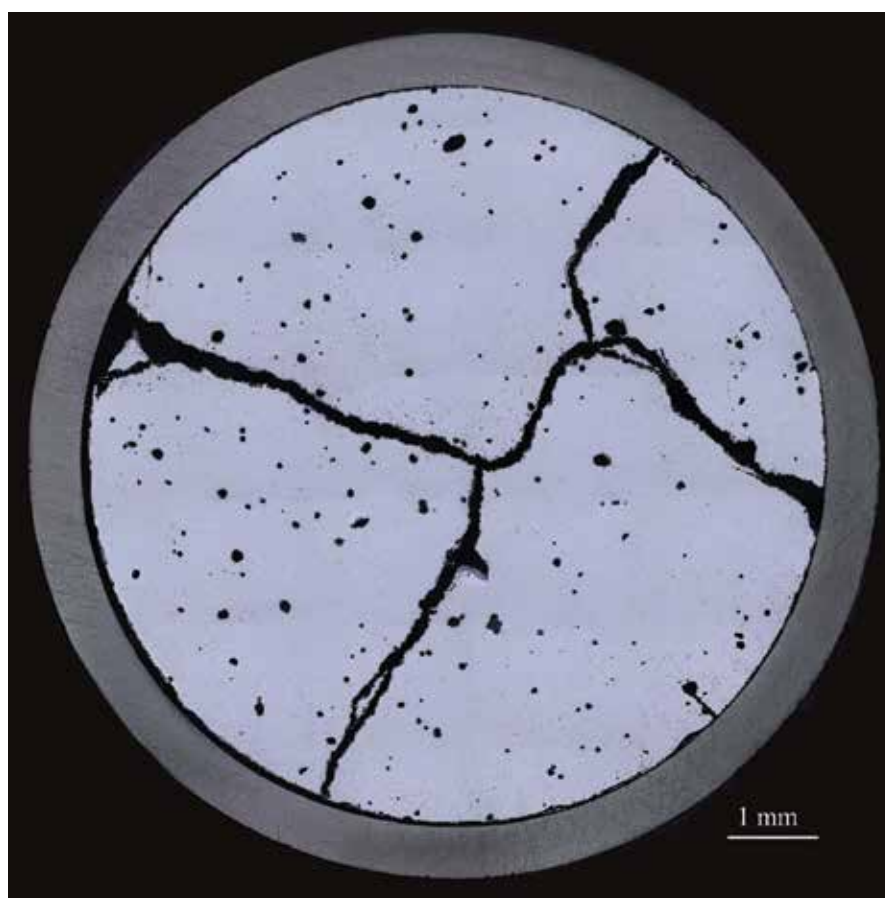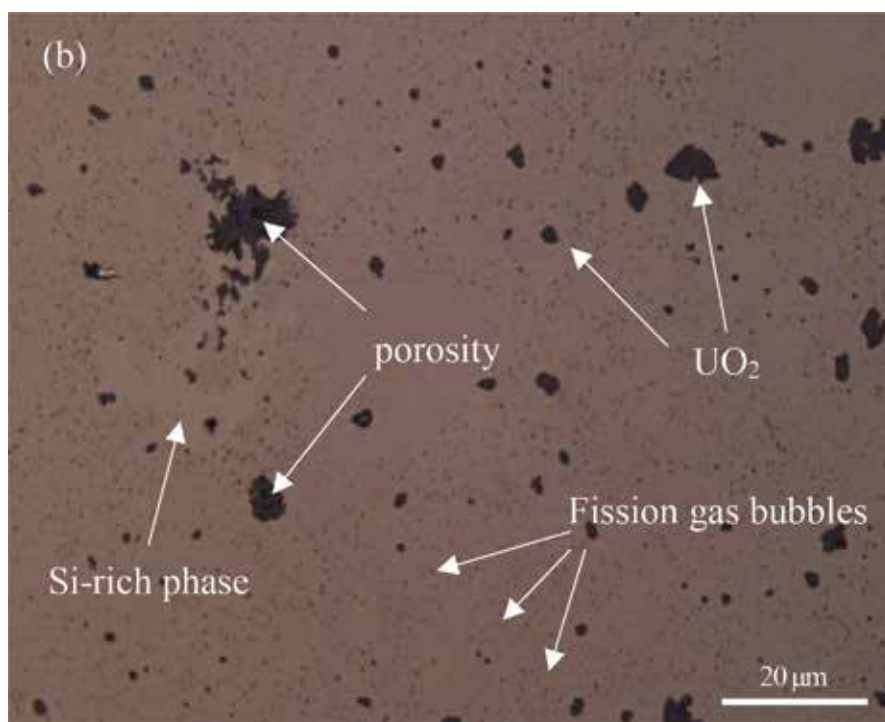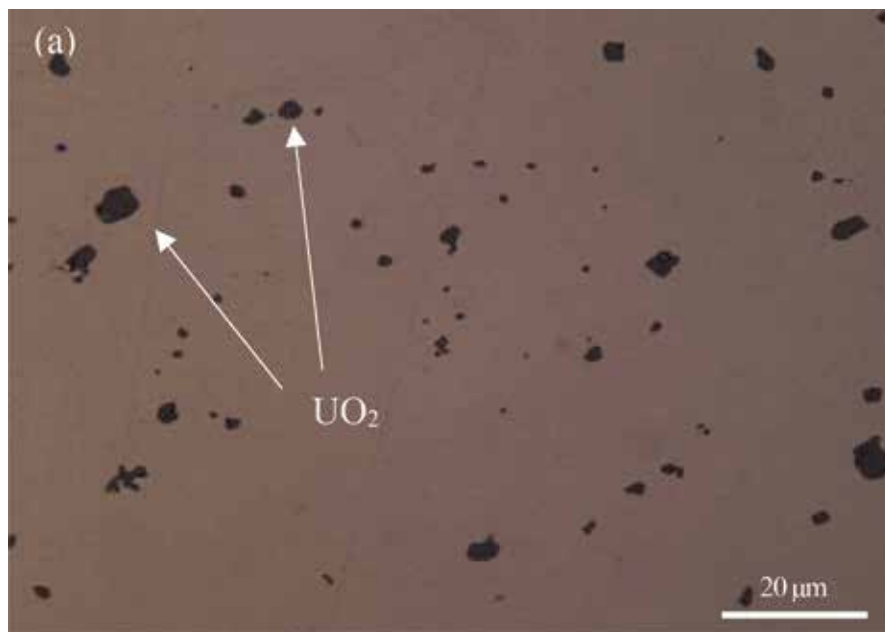Figure 1. Thermal neutron radiography image of the two rodlets.

Figure 2. Low magnification (50X) ceramography cross section of irradiated U_3Si_2.

Figure 3. Representative high magnification (100X) optical microscopy images of the fuel pellet microstructure taken from (a) pellet periphery and (b) pellet center.

Accomplishments:

The PIEs took place in the Hot Fuel Examination Facility (HFEF) at INL. A broad spectrum of PIE has been employed to evaluate fuel integrity, fission gas release, fission product distribution, burnup, fuel swelling and cladding strain. Performed PIE included neutron radiography, gamma spectrometry and tomography, dimensional inspection, fission gas release and chemical burnup analysis. Microscopy samples were examined using the new high resolution optical microscope (OM) installed in HFEF.

Analysis of neutron radiography images (Figure 1) and low magnification OM images (Figure 2) show a limited number of cracks in the fuel matrix, as direct consequence of the reduced thermal stresses. Figure 3 shows two examples of the irradiated U_3Si_2 microstructure, one acquired on the pellet periphery (Figure 3a) and the other at the center (Figure 3b). As in the as-fabricated microstructure, secondary light and dark phases are present in the main matrix, corresponding to Si-rich and UO_2 impurities present from fabrication. Porosity is co-localized with the

UO_2 phase. Towards the pellet center (Figure 3b), additional small spherical features appear, which are likely to be fission gas bubbles. Image analysis algorithms were developed to perform segmentation of the secondary phases and fission gas bubbles. The value of the porosity associated with the UO_2 precipitates remains as in the as-fabricated samples, excluding densification. Fission gas bubbles development is consistently observed within 60% of the pellet radius for both irradiated rodlets. The formation of bubbles in the central part of the pellet is consistent with the local higher irradiation temperatures. It is not to be excluded that smaller bubbles not resolvable with the optical microscopy exist in the pellet periphery. However, the overall swelling of the pellets remains limited and fission gas release remains very low (0.05% of the produced inventory).

The microstructure investigation and the overall results of the PIE campaign suggest a good performance of this ATF candidate at low burnup and under operating conditions.

Development of Characterization Methods for Post-irradiation Examination of MiniFuel Geometries

Principal Investigator: Alicia M. Raftery
Collaborators: Robert N. Morris, Kurt R. Smith, Grant W. Helmreich, Christian M. Petrie, Kurt A. Terrani, Andrew T. Nelson

Figure 1. Cutting fixture shown loaded with a MiniFuel test sub-capsule before cutting (left) and capped for transfer after cutting (right).

Irradiation testing of fuel is highly recognized as an essential aspect of nuclear fuel research and development. The traditional approach for fuel testing typically involves full-scale integral irradiation campaigns that can require extensive use of funding and resources. While this approach is necessary for more mature concepts, early stage testing of novel fuel concepts may prove beneficial by highlighting poor or unexpected irradiation behavior before it reaches the stage of full-scale testing. The Miniature Fuel (MiniFuel) concept was designed at Oak Ridge National Laboratory (ORNL) to address this need by providing a resource for screening, benchmark testing, and separate effects screening of fuels by irradiating small fuel geometries in the High Flux Isotope Reactor (HFIR). The approach uses small samples and a unique irradiation vehicle to accumulate burnup under highly controlled conditions. This greatly reduces cost while increasing control of numerous variables that cannot be easily accessed using conventional integral irradiation testing.

Project Description:

The MiniFuel concept uses a state-of-art design to irradiate very small fuel materials, with volumes on the order of a few cubic millimeters; currently planned geometries range from 3mm diameter disks to as small as 0.425mm diameter microspheres. This approach enables screening tests on novel fuel

Figure 2. Fission gas puncture unit shown during testing (top left and right) and puncture tool shown after successful puncture with minimal dulling of tip (bottom).

concepts, which could highlight irregular irradiation behavior much earlier than the traditional approach of using full-scale irradiation campaigns. In addition, the sub-capsule design and small specimen size allow for separate effects testing of various irradiation parameters that are otherwise difficult to obtain using full-size designs. For example, the small specimen size removes temperature gradients that are present in larger fuel specimens during irradiation. Separate effects data on currently existing fuels retrieved as part of the MiniFuel

project is expected to contribute as input data to fuel performance codes. While the small specimen size has advantages in regard to separate effects testing, it also creates challenges for post-irradiation disassembly and examination. Significant development and adaptation of previously established methods was necessary to ensure the successful retrieval of fuel specimens and the collection of meaningful data. The first set of MiniFuel irradiations, which included 0.8mm uranium nitride microspheres and 1mm tri-structural isotropic (TRISO)

Development of novel PIE methods optimized for miniature fuel volumes as used by the MiniFuel project will provide valuable screening and separate effects data on currently existing and novel nuclear fuels.

particles, were loaded into HFIR in the summer of 2018 and will be ready for postirradiation examination (PIE) by the end of the year. A major emphasis of this project has been development of the PIE methods that will be used to obtain results on the irradiation behavior of these and future MiniFuel specimens.

Accomplishments:

The procedure at ORNL for disassembly, fuel specimen retrieval, post irradiation analysis of the fission product behavior, and microstructural characterization of the MiniFuel specimens has been established. The process for disassembly was formulated to ensure retrieval of the small specimens given the visual limitations that exist in various cells at the Irradiated Fuels and Examination Laboratory (IFEL) hot cell facility. The cutting procedure includes cutting the sub-capsule while in a vertical orientation, capping the sub-capsule, and transferring it to a cell which is equipped with a vacuum needle handling tool and a stereo micro-

scope for fuel retrieval. A cutting fixture and cap have been designed specifically for use with the MiniFuel design and are shown in Figure 1. The cutting methods have been tested and verified as capable of ensuring the specimens are kept in-tact until final retrieval.

Fission gas release is an important parameter used to analyze the fission product behavior in irradiated fuels. A fission gas puncture system, which was previously used on full-length fuel rods at ORNL, has been adapted for use with the MiniFuel design to measure the fission gas release of the specimens. The main modification consisted of designing a gas puncture unit with the dedicated purpose of puncturing the MiniFuel sub-capsule. Figure 1 shows the gas puncture unit out-of-cell during testing and the puncture tool, which has a 45° heat treated A2 steel tip that is screw-driven into the sub-capsule until puncture occurs. The gas puncture unit and flow system functions have been demonstrated successfully on un-irradiated sub-capsules.
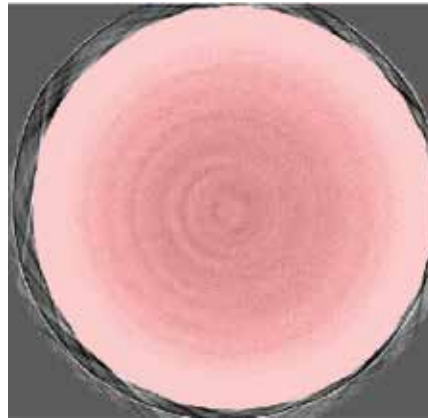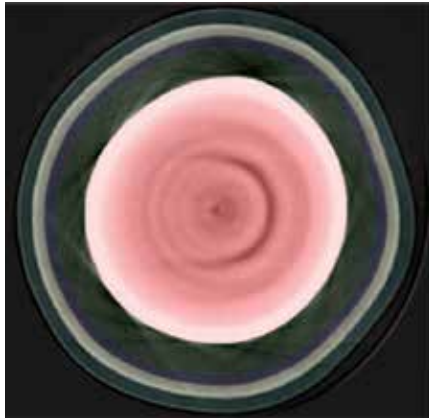
Figure 3. Cross-section of an un-irradiated UN TRISO tomograph (left) and cross-section of a tungsten carbide standard tomograph (right) with coloration of segmented layers.

The dimensional change of the fuel as a result of irradiation also gives insight into the fission product behavior. A MATLAB program for determining the post-irradiation volume of the fuel specimens using x-ray computed tomography (XCT) tomograph images has been created and validated using standards of similar size to the MiniFuel specimens. Figure 3 shows the cross-section of a uranium nitride TRISO tomograph and a tungsten carbide kernel tomograph with coloration of segmented layers, which is used in the volumetric analysis.

The methods for microstructural characterization of MiniFuel are based on past experience with particle fuel development at ORNL and include mounting microspheres for polishing in-cell and subsequent analysis with a variety of electron microscopes located at IFEL and the Low Activation Materials Development and Analysis (LAMDA) laboratory. These methods are directly applicable for use with the small geometries expected for Mini-Fuel irradiations.

All methods have been demonstrated and are ready to be implemented in-cell and used on the first set of MiniFuel specimens, which will provide process experience for further development. Future work on post-irradiation examination methods for MiniFuel includes the development of additional methods to analyze thermophysical and mechanical properties.

Separate Effects "Mini Fuel" Irradiation Testing

Principle Investigator: Christian Petrie
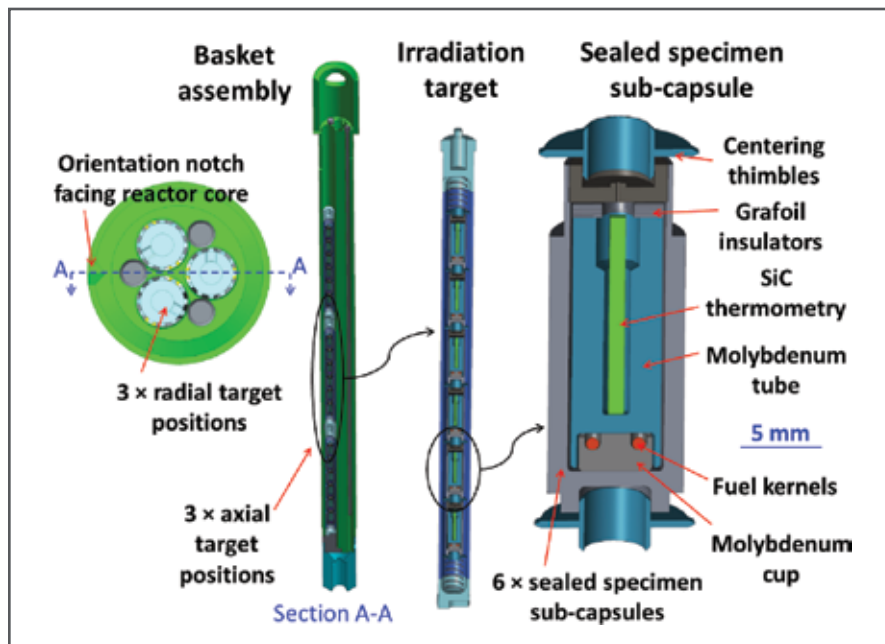*Collaborators: Joseph Burns, Annabelle Le Coq, Alicia Raftery, Jake McMurray, Robert Morris, Kurt Terrani*

Figure 1. Mini fuel design concept.

The ability to perform well-controlled separate effects irradiation testing would allow for evaluation of a wide range of new fuel concepts within a reasonable time and cost.

This goal of this project is to implement a new capability for performing accelerated separate effects irradiation testing of miniature ("mini") fuel specimens in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). The design concept places the mini fuel specimens inside individually sealed sub-capsules inside steel targets in the reflector of the reactor. Temperature is controlled by sizing an insulating gas gap between the sub-capsules and the target housing. Reducing the size of the fuel allows for very high fission rates (on a per unit mass basis) without prohibitively large temperature gradients. Furthermore, the small fuel mass results in the total heat generated in each sub-capsule being dominated by gamma heating in the structure instead of fission in the fuel itself. This essentially decouples the fuel temperature from the fission rate.

The benefit of this work to the nuclear community is that advanced fuel concepts can now be evaluated early in their development process under very controlled conditions, and in a much more economical manor. Using the mini fuel capability, several fuel performance variables can be explored under a single irradiation campaign including temperature, burnup, composition, geometry, enrichment, grain size, impurities, and non-stoichiometries. Ultimately this

Qualification of new nuclear fuel materials requires a thorough scientific understanding of fuel behavior including irradiation performance. Traditionally, irradiation performance data has been acquired through many integral fuel irradiation experiments where full-size fuel pellets are tested under conditions that closely match those of the intended application. While this approach is logical, it is very expensive and time consuming, particularly when considering a large test matrix. Furthermore, the large number of variables that affect fuel performance make it difficult to develop fundamental models of various phenomena from integral fuel tests, which often have many independent variables that cannot be well-controlled.
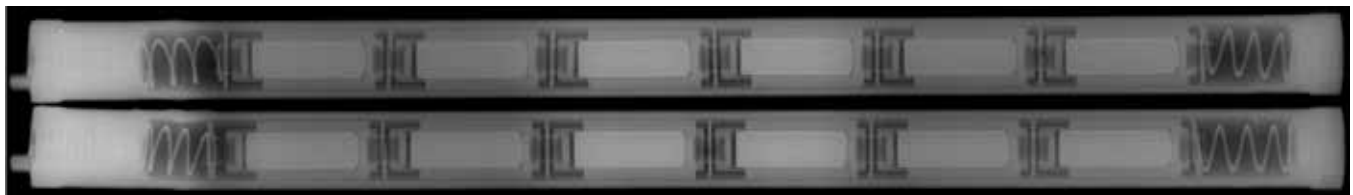
Figure 2. Radiograph of welded targets prior to irradiation.

capability will help deliver advanced fuel forms that offer enhanced performance and improve the safety and economics of nuclear power production.

The first mini fuel experiments have been inserted into the HFIR starting in cycle 480 (June 2018). This first set of experiments is investigating irradiation effects on uranium mononitride (UN) kernels with a nominal diameter of 800 μm. A wide variety of kernels have been included in the testing including variations in kernel density (87-95% of theoretical density), carbon impurities, and burnable absorbers (Gd). The fuel kernels were fabricated at ORNL using a sol-gel process. Tristrictural isotropic (TRISO) coated particle fuel utilizing the same UN kernels was also included in the test matrix. The design temperature for these irradiations is 550°C, which is relevant for light water reactor applications. Higher temperatures can be achieved using a more insulating fill gas. One set of fueled capsules is targeting a lower burnup of 1% fission of initial metal atoms (FIMA). A second set of capsules will be irradiated to a target burnup of 6% FIMA.

All fuel kernels use either depleted or natural uranium. Despite the lower enrichment, very high particle powers (on the order of 500 mW, or 150 W/g UN) can be achieved due to the extremely high neutron flux in the HFIR. Neutron absorption in 238U results in rapid breeding of 239Pu. Eventually, an equilibrium between 239Pu production and fission is achieved. At this point, the fuel fission rate remains nearly constant for the remainder of the irradiation. This greatly reduces the effects of fuel burnup on the fuel temperature over the course of the irradiation. In addition, the small size of the fuel naturally minimizes the sensitivity of the fuel temperature to fluctuations in fission rate (i.e., the total experiment heating is dominated by gamma heating in the structure).

After irradiation, the experiments will be shipped to the Irradiated Fuels Examination Laboratory where the experiment targets will be cut open and the individual capsules will be extracted. Fission gas release will be measured for the fuel contained inside each capsule via puncturing and cold trapping of xenon and krypton isotopes. Swelling measurements will be performed using either gas pycnometry or X-ray computed tomography. Finally, microstructural characterization will be performed to evaluate irradiation effects on the kernels and (for the TRISO particles) the coating layers.

The Sensor Qualification Test

Principal Investigator: Gary Povirk
Collaborators: Bryon Curnutt, Brian Durtschi, Stephen Evans, Gary Hoggard, Nate Oldham, Connor Woolum

Top Tier Loaded with Sensors to be Qualified   Bottom Tier Loaded with "Dummy" Pins

Figure 1. A schematic of the instrumented top tier of the Sensor Qualification Test and the bottom tier with uninstrumented pins.

Improved on-line instrumentation for fuel system parameters such as temperature, axial extension of the fuel, and gas release offer the potential to speed development through improved understanding of in-reactor phenomena. Typically this information either cannot be obtained at all (e.g., fuel temperatures) or can only be acquired through destructive post-irradiation examinations. In the latter case, destructive examinations require a substantial amount of additional resources and time to complete and the limited number of examinations that can be performed provide only a partial picture of in-reactor behavior. The objective of the Sensor Qualification Test was to demonstrate the ability to reliably obtain data from the loop in the center flux trap of the Advanced Test Reactor (ATR), with the ultimate goal being the instrumentation of fuel pins for the Accident Tolerant Fuel-2 (ATF-2) test.

Project Description:

The Sensor Qualification Test was a non-fueled experiment intended to demonstrate the capability to instrument the loop in the center flux trap of the ATR. The experiments were intended to mimic the instrumentation planned for the ATF-2 test, with thermocouples for temperature measurements as well as linear variable displacement transducers (LVDTs) for measurements of axial fuel growth and gas release. In the case of gas release, the LVDT measured the deflection of a small bellows, which in turn was a function of the gas pressure within the pin. Given that the tests were non-fueled, a gas panel was designed so that a pressure history within the rod could be prescribed and in turn enable comparisons between the applied pressure and the pressure measured from the displacement of the bellows. In addition, the rodlets were fabricated with tungsten pellets which served as a source of gamma heating to elevate temperatures within the pin. Figure 1 shows a schematic of the experiment and of the instrumentation used in the test.

Accomplishments:

The Sensor Qualification Test was substantially more complex than a typical non-fuel experiment in the ATR. Test execution required the design and fabrication of instrumentation that was housed within dummy fuel rods, running gas lines and electrical leads out of the test train, through the reactor head, and into the gas panel and the data collection systems. An experiment in the ATR critical facility was required to ascertain the impact of the equipment on ATR reactor physics

and shake-down testing was performed at the Westinghouse flowing autoclave facility outside of Pittsburgh to assess the reliability of the instrumentation. Based on the extraordinary efforts of the all involved, the Sensor Qualification Test was inserted into the ATR in September of 2017. Figure 2 shows the team alongside the completed test train, just prior to insertion.

Unfortunately, a few days after the test began, tungsten was detected in the loop coolant which indicated that at least one of the instrumented pins was breached. Although there is no conclusive evidence as to the cause of the breach, the most likely explanation is judged to be failure of a thermocouple sheath. In the first insertion of the ATF-2 experiment in June of 2018, no instrumentation was provide based in part because of the suspected thermo-

couple failures in the Sensor Qualification Test, but also because of schedule pressures and because difficulties in the development of a particular ATF concept reduced the need for instrumentation. In spite of these challenges, however, the Sensor Qualification Test has made and will continue to make a lasting impact in our ability to develop and test advanced light water reactor fuel systems. In the near-term, the test demonstrated that the instrumentation technologies were not sufficiently mature to be used in fueled test experiments so that it likely prevented fuel pin leaks in the ATF-2 test. Moreover, the infrastructure that was put into place in the ATR as part of this test will undoubtedly be used in future experiments when the reliability of the instrumentation has been more clearly demonstrated.

Supporting ATF Fuel Vendors: Fabrication of Rodlets

Principal Investigator: Connor Woolum
Collaborators: Brian Durtschi, Nate Oldham, Chris Murdock, Kyle Kofford, Kip Richards

Figure 1. Laser welding of ATF-2 fuel rodlet.

The ATF-2 experiment includes irradiation of accident tolerant fuel (ATF) concepts in prototypic pressurized water reactor (PWR) conditions within the Advanced Test Reactor (ATR). One of the many aspects of this complex experiment is the fabrication, assembly, and qualification of irradiation test specimens, both fueled and non-fueled. Additionally, specimens fabricated externally require proper receipt and evaluation to verify all requirements are met both programmatically and for reactor safety purposes

Project Description:
Irradiation testing of ATF concepts in PWR conditions provides valuable information to fuel vendors to support future research endeavors and licensing efforts for the ATF concepts. The initial insertion of the ATF-2 test train occurred in June of 2018 and was comprised of a variety of novel fuel and/or cladding concepts from multiple industry partners. The test train is a 5-tiered modular design, each tier designed to hold up to 6 fuel rodlets. Each fuel rodlet is nominally 6" or 12" long and made of an experimental cladding and fuel or non-fuel internal components.

Figure 2. Setting up for fuel rodlet welding within the laser welder.

In addition to the novel ATF concepts, the first insertion of the ATF-2 test train included several fuel rodlets with standard UO2 fuel and Zr-4 cladding to serve as reference specimens for future irradiation testing in Idaho National LaboratoryINL's transient test reactor (TREAT). Some of these UO_2/ Zr-4 fuel rodlets may also serve as standard points of comparison with ATF concepts during postirradiation examination (PIE), given the well characterized performance of this fuel system. INL fabricated a majority of the test specimens, while others were provided by industry collaborators and qualified for reactor insertion at INL

Accomplishments:

The ATF-2 campaign set ambitious goals with the first insertion of the test train both with the quantity and with the variety of fuel rodlets. In total INL was responsible for, and successful in, the fabrication and assembly of fifteen rodlets and receipt and qualification of an additional eleven rodlets.

Building upon lessons learned from previous irradiation experiment fabrication efforts, the assembly of ATF-2 fuel rodlets employed laser welding as a more reliable and robust assembly method than convention TIG welding

Figure 3. ATF-2 fuel rodlet.

as applied to these types of designs. A new laser welder was commissioned and assembly methodology developed to support the high yield assembly campaigns required as part of the ATF-2 experiment.

In addition to the laser welder for circumferential welds on the rodlets, a TIG welding system was designed, built, and deployed in order to perform the final rodlet weep hole weld. This new TIG welding system, called the Welding Under Pressure System (WUPS), proved very effective in performing the final closure weld for the fuel rodlets. WUPS allowed for rodlet internal fill gas composition and pressure to be tailored to meet programmatic requirements.

These improvements in both equipment and technique resulted in a 100% yield for every fuel rodlet build campaign; that is, every weld passed requisite inspections without any rework required.

Eleven more rodlets were fabricated by an industry partner and received by INL. The receipt process included a

Figure 4. Radiographic image of ATF-2 fuel rodlet.

Figure 5. Fuel pellets being loaded into ATF-2 fuel rodlet.

thorough review of as-built information and inspections as performed by the partner. Additionally, some inspections were performed on-site at INL as verification of critical characteristics to ensure both programmatic and safety requirements were met.

Twenty-six test specimens, including fifteen assembled by INL, were qualified for irradiation testing within the 2A loop of ATR in support of the ATF-2 experiment.

2.6 Fuel Safety Testing

Fuel Safety Testing – TREAT (Advanced LWR)

Principal Investigator: Dan Wachs
Collaboration: Todd Pavey

> **Fuel safety research is a vital part to determine improved performance, reliability, and safety characteristics of Accident Tolerant Fuels for Light Water Reactors.**

The Transient Reactor Test (TREAT) Fuel Safety Testing for Advanced Light Water Reactor (LWR) has continued with efforts in improving Accident Tolerant Fuels (ATFs), transient prescription testing, developing separate effects testing plans, and conducting transient prescription modeling to validate TREAT's ability to execute fuel safety research relevant to both overpower and undercooling type transients. The transient test program has developed an inexpensive static dry capsule with a reusable primary containment to run the first fueled transient test at TREAT in September 2018 since TREAT was placed in standby in 1994.

Project Description:

The objective of the TREAT Fuel Safety Research for advanced LWR is to develop Accident Tolerant Fuel for the next generation of LWR fuels with improved performance, reliability, and safety characteristics during normal operations, accident conditions and with reduced waste generation. Fuels with enhanced accident tolerance are those that can tolerate loss of active cooling in the core for a considerably longer time period than the UO_2 zircaloy system while maintaining or improving the fuel performance during normal operations, operational transients, and design-basis events. The Nuclear Science and Technology (NS&T) Transient Testing program is working on developing modular capsules that can be used to conduct transient testing experiment campaigns to test different fuel types. A dry static capsule has been developed to establish a baseline for future fueled experiments allowing a baseline comparison of data. The dry capsule is also used to conduct transient testing for separate effects testing (SET), which will allow universities to work with TREAT to perform different fueled SET experiments. NS&T is designing the static wet capsule, Minimal Activation Retrievable Capsule Holder (MARCH)-Static Environment Transient Testing Apparatus (SERTTA) to primarily focus on Reactivity Insertion Accident scenarios, although it will also be used to perform a less expensive and less detail Loss of Coolant Accident (LOCA) test. NS&T is designing a natural circulation flow water capsule (Super-SERTTA) to primarily focus on LOCA testing with different fuel types, although this capsule will be able to be used for Reactivity-Initiated Accident (RIA) testing also. The transient testing program worked with TREAT operations to conduct several transient prescriptions to obtain the necessary information to properly prescribe the transients required for several series of tests. NS&T transient testing program also conducted transient prescription modeling to validate TREAT's ability to execute fuel safety research relevant to both RIA and LOCA type transients.

Accomplishments:

Completed ATF 3-1 Transient Prescriptions per Data Package DP-125

The purpose of the ATF-3-1 transient prescription tests were to obtain the necessary information to properly prescribe the transients required to conduct the ATF-3-1 series of tests. Additional outcomes include establishing a basis for the range of repeatability of core energy release based on uncertainty of reactivity insertions, particularly during clipped transients, developing TREAT operational knowledge, and gathering crucial data for nuclear/kinetic model comparisons for ATF-3-1 transients. This was completed by Nicolas Woolstenhulme, David Kamerman, Dan Wachs and the TREAT Operations team.

Completed Narrow Pulse Width Transients per Data Package DP-126

The purpose of the narrow pulse width transients is to exhibit the ability of the TREAT facility to perform large reactivity insertions with repeatable timing of clipping to shorten the pulse width as compared to a temperature limited transient (TLT). As part of the Accident Tolerant Fuel (ATF) program and TREAT's ability to support Light Water Reactor (LWR) testing in general, it is desired for the pulse width to be made shorter than 100 milliseconds (ms), as defined by the Full-Width Half-Maximum (FWHM), a typical metric for measuring length of a pulse. This

was completed by Nicolas Woolstenhulme, David Kamerman, Dan Wachs and the TREAT Operations team.

Completed the LOCA Transient Prescription Study per Data Package DP-128

The purpose of the present data package is to describe shaped transients with relevance to Loss of Coolant Accident (LOCA) simulation capabilities in the Transient Reactor Test facility (TREAT). The objective of these transients is to demonstrate that TREAT can perform long-duration shaped transients for input to concurrent experiment designs including capsules and loops with the ability to create LOCA-like thermal hydraulic boundary conditions for fuel specimens. This was completed by Nicolas Woolstenhulme, David Kamerman, Dan Wachs and the TREAT Operations team.

Completed a critical milestone, completed the first two fueled experiment at TREAT since it was restarted. The test was completed on fresh fuel in a SETH container. This was completed by Nicolas Woolstenhulme, Clint Baker, Benjamin Baker, Devin Imholte, Colby Jensen, Dan Wachs and the TREAT Operations team.

Commenced conceptual design of MARCH-SERTTA and Super-SERTTA capsules for future testing. This is being worked by Nicolas Woolstenhulme, Clint Baker, Nate Oldham, Colby Jensen, and Dan Wachs.

Re-fabrication of Irradiated Fuel Rods and LOCA Burst Testing

Principal Investigator: Kory Linton
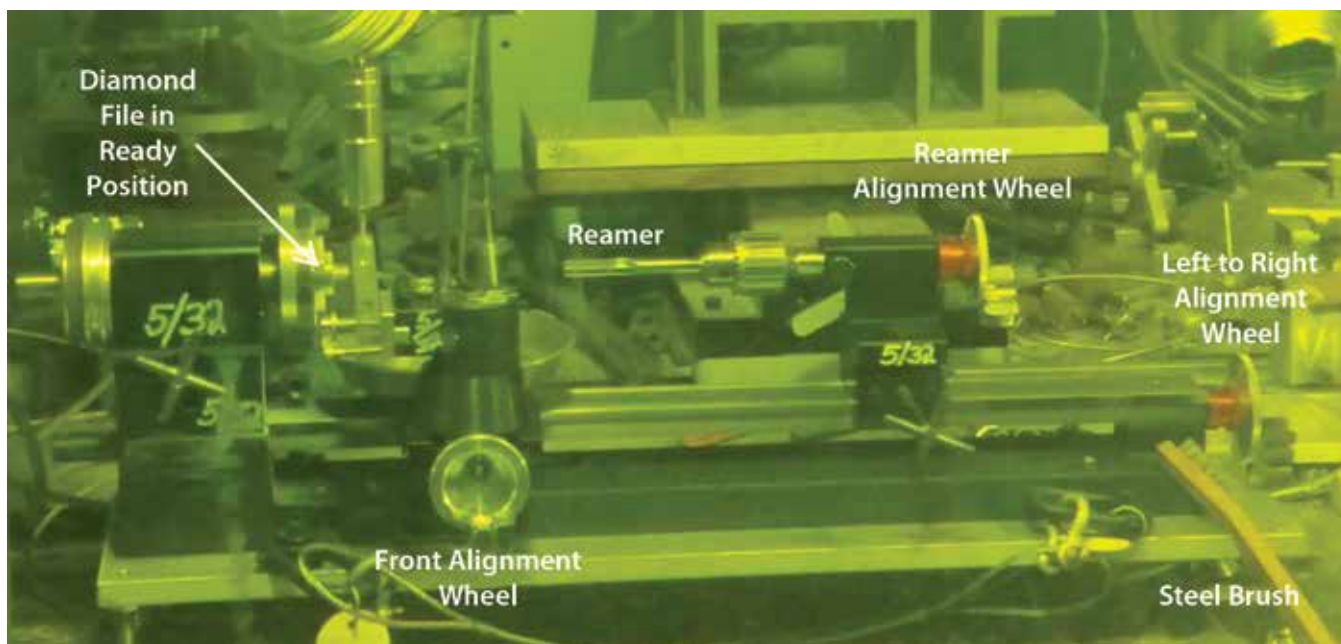*Collaborators: Yong Yan, Zach Burns, Tyler Smith, Alicia Raftery, Kurt Terrani*

Figure 1. Lathe assembly used for removal of the oxide layers on the OD (Diamond File) and ID (Reamer).

A high burnup commercial spent fuel segment was re-fabricated into a pressurized test train for fuel safety testing in the Irradiated Fuels and Examination Laboratory (IFEL) hot cell facility at Oak Ridge National Laboratory (ORNL). The in-cell integral Loss of Coolant Accident (LOCA) test was performed in the Severe Accident Test Station (SATS) by subjecting the spent fuel segment to a steam environment, internal pressure, temperatures up to 1200°C, and water quench. The successful in-cell re-fabrication and integral LOCA demonstration test restores a domestic capability for out-of-pile integral fuel safety testing. In addition, this work establishes the SATS system as a vital tool for evaluation of commercial spent fuel and future accident tolerant fuel (ATF) concepts subject to design basis accident (DBA) and beyond design basis accident (BDBA) scenarios.

Building on the 2017 successful in-cell installation and demonstration of the Severe Accident Test Station, the two main objectives for this project included: (1) establishing a procedure for in-cell re-fabrication

of LOCA test trains and (2) demonstrating this capability by performing an in-cell integral LOCA test on high burnup commercial spent fuel.

To perform LOCA tests on irradiated spent fuel, the fuel must be refabricated into test trains utilizing dedicated hot cell facilities capable of fuel rod sectioning, preparing metallographic mounts, fuel leaching, removing oxide layers from cladding, and fitting end-plugs either by welding or Swagelok. Fuel leaching is performed to remove approximately .5" of fuel from the ends of the fuel segment by using a beaker filled with nitric acid and hot plate. Metallographic mounts are prepared to determine the oxide layer thickness on the inner and outer diameter of the cladding. The measured oxide layer on the cladding end plug region is then removed by reaming the inner diameter and using a diamond file on the outer diameter using a lathe assembly equipped with both devices (Figure 1). Each of these steps helps to ensure an effective seal between the end-plugs and cladding during the welding process or when using a Swagelok fitting. Final assembly of the test train includes wrapping thermocouples 2" above and below the midplane, adding pressure lines and centering rings and inserting the assembled experiment into a quartz tube.
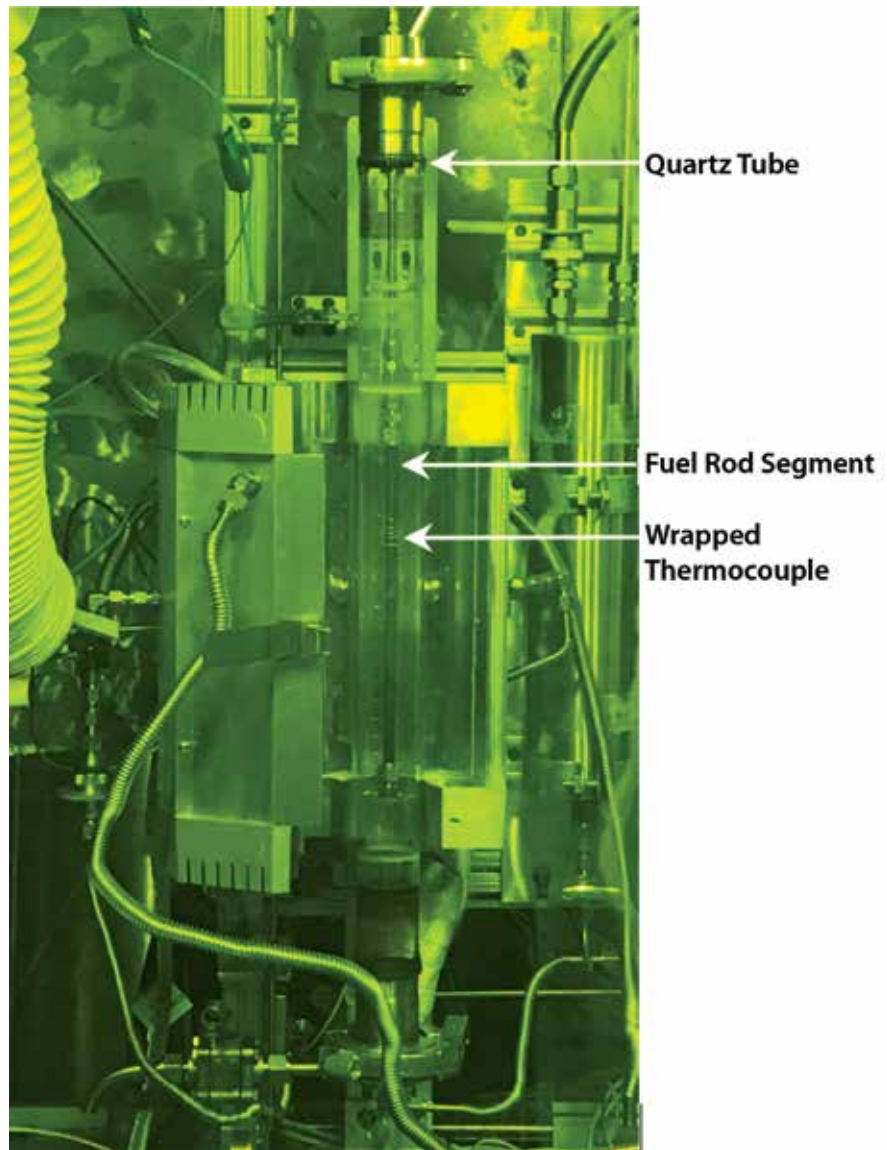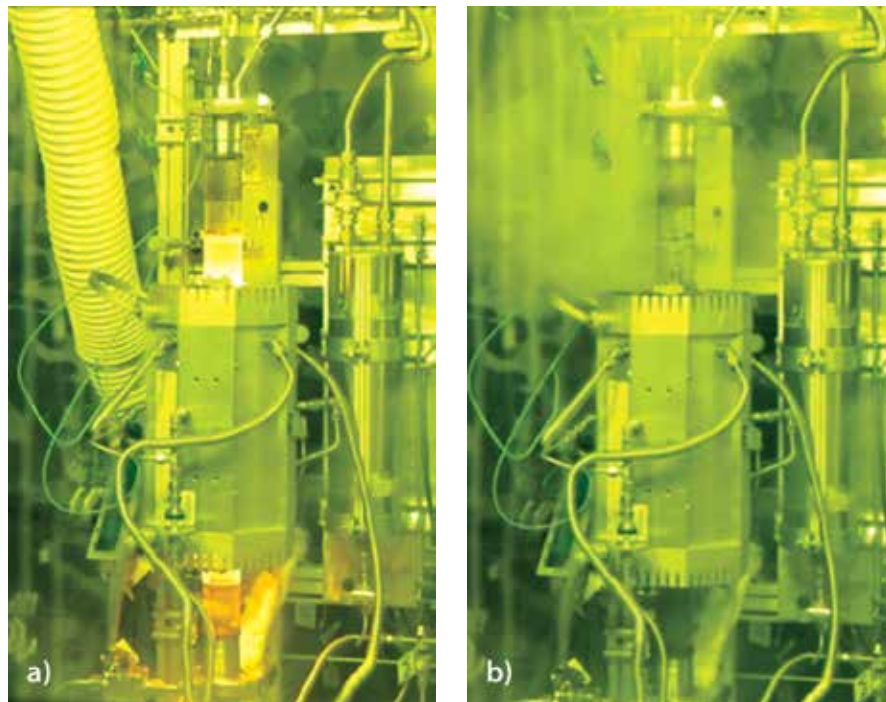

Figure 2. North Anna fuel segment loaded in the SATS LOCA test apparatus

Figure 3. Photographs taken during in-cell LOCA test showing (a) the temperature ramp and (b) immediately following water quench

Moving forward, in-cell fuel safety research will extend to post LOCA mechanical testing and microstructural characterization. In parallel, testing of ATF concepts using SATS will require development of shipping pathways for ATF-1 rodlets from INL to ORNL, planning for shipping and post irradiation examination of Hatch-1 FeCrAl rodlets at ORNL, and exploration of alternate test geometries to accommodate shorter segments of High Flux Isotope Reactor (HFIR) irradiated cladding geometries.

The SATS LOCA test apparatus for this experiment (Figure 2) includes the North Anna segment inside of a quartz tube and a fully instrumented test train. The test train has the capability of measuring the temperature and pressure throughout the experiment. The test was conducted with two Type-S thermocouples strapped to the outer surface of the cladding approximately 2 inches above the sample centerline. One of these was used to control the furnace power to give a hold temperature of 1200°C at that location. The other

> *In-cell re-fabrication of a high burnup commercial fuel segment into an integral LOCA test train and successful testing in the Severe Accident Test Station demonstrate the readiness to evaluate accident tolerant fuel (ATF) concepts subject to design basis accident (DBA) and beyond design basis accident (BDBA) scenarios.*
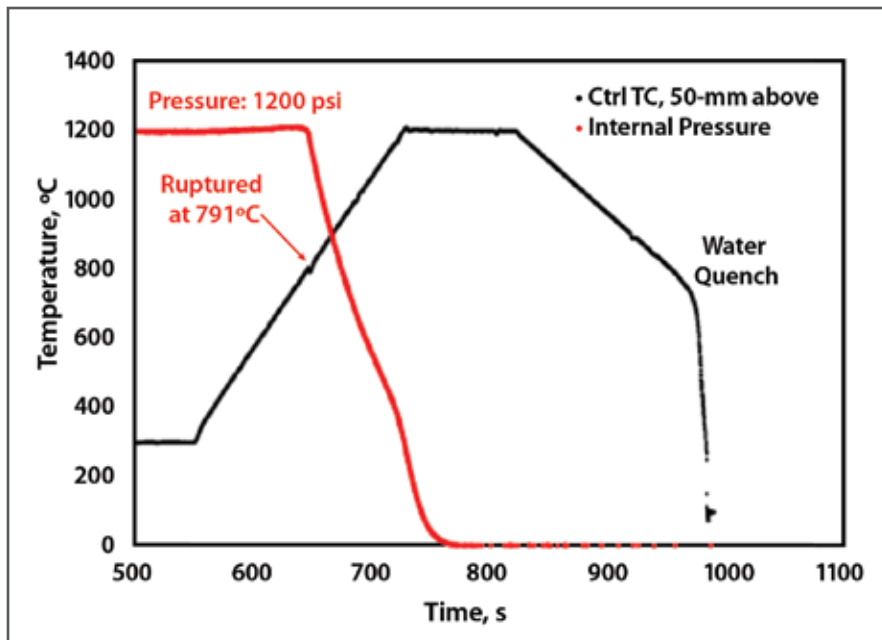

Figure 4. Temperature and pressure histories for the in-cell LOCA test NA#1 through full integral test sequence, including water quench at 800°C

Table 1. Comparison of the
ORNL NA#1 LOCA test and
ANL Limerick test under the
same temperature, pressure,
and quench test conditions

Parameter	ANL ICL#4	ORNL NA#1
Fuel	Limerick	North Anna
Materials	Zircaloy-2	M5
Burnup, GWd/MTU	54-57	67.3
OD, mm	11.18	9.50
Wall Thickness, mm	0.71	0.57
Test Environment	Steam	Steam
Hold Temperature, ºC	1204	1200
Hold Time, s	300	90
Quench Temperature, ºC	800	800
Max. Pressure, psi	1285	1214
Burst Pressure, psi	1160	1189
Burst Temperature, ºC	≈790	≈791
Burst Shape	Oval	Oval
Burst Length (L_b), mm	15	16
Max. Burst Width (W_b), mm	5.1	3
Balloon Length (L_{bal}), mm	≈80	≈50
Max $\Delta D/D_0$, %	36 ± 10	41 ± 2
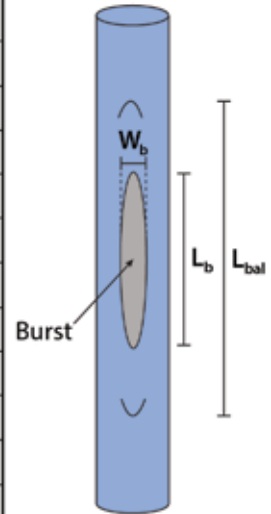
thermocouple, located 180ºC from the control thermocouple, was used as a backup thermocouple to record temperature in the event that the control thermocouple failed.

The experiment was conducted as a fully integral test, meaning multiple out-of-cell phenomena expected to occur during a LOCA were simulated. The full LOCA sequence included:

- Heating in flowing steam to 300ºC and pressurizing fuel segment to 1200 psi
- Heating in flowing steam at 5ºC/s from 300ºC to 1200ºC
- Holding in steam for 90s at 1200ºC
- Cooling at 3ºC/s
- Rapid cooling by water quench at 800ºC
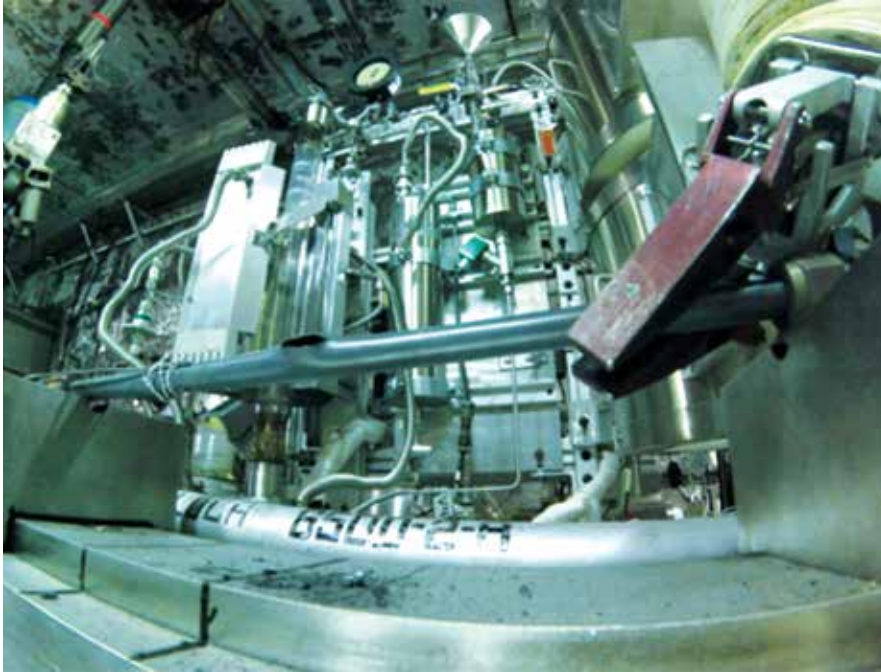
Figure 5. Manipulator holding the burst NA#1 fuel rod segment in front of the SATS after the in-cell LOCA test

Figure 3(a) shows an image taken during the heating step of the test and Figure 3(b) shows an image taken during the water quench.

The temperature and pressure evolution throughout the entire LOCA test, including the quench, is shown in Figure 4. The results from this experiment were compared to a LOCA test that was conducted at Argonne National Laboratory (ANL) in order to validate the operation and behavior of the SATS system. Table 1 summarizes the results for the ANL integral LOCA test with high-burnup fueled Zr-2 cladding and the current ORNL in-cell test of M5 cladding, referred to as NA#1. The ORNL in-cell test was run with a full LOCA sequence with the same pressure, temperature, and quench test conditions of ANL ICL#4, except for a shorter hold time. Although the sample size and cladding material are different between the two tests, the burst temperature and pressure of the ORNL NA#1 test closely follow the ANL ICL#4 test, which is shown by the similar values for the test results. The burst in the cladding and some fuel fragmentation behavior is pictured in Figure 5.

Metallic Americium Feedstock Production

Principal Investigator: K.Robb
Collaborators: K. Terrani, A. Nelson, Micro Große (KIT), Martin Steinbrueck (KIT)

Robb_1 Temperature, power, and hydrogen generation data collected during QUENCH-19 test of FeCrAl test bundle compared with data collected for ZIRLO during a previous test.
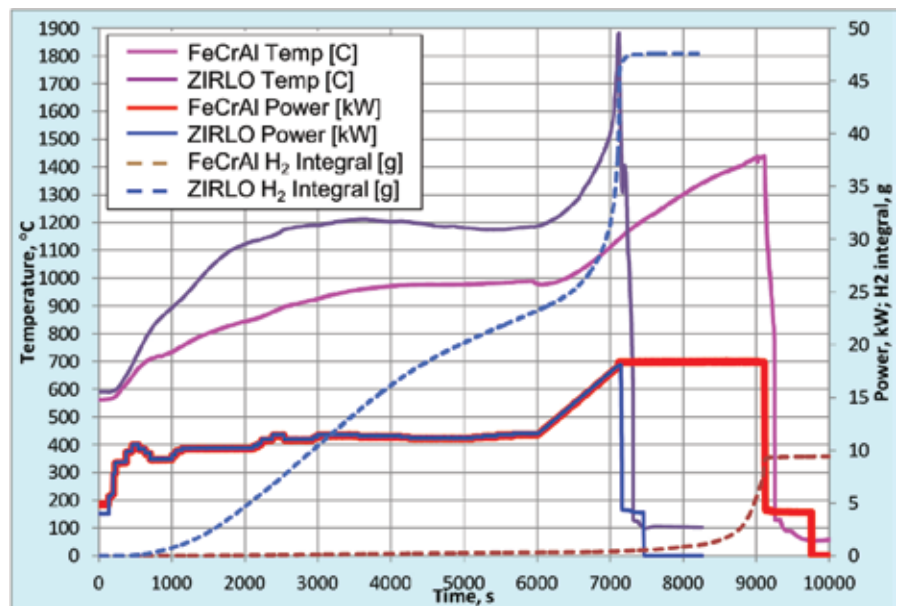
Response of accident tolerant fuel (ATF) cladding alloys to a loss-of-coolant accident (LOCA) is a critical metric in their assessment. Cladding concepts are initially tested as coupons or small individual samples. If acceptable performance is observed, testing then extends to tube geometries to examine behaviors more relevant to reactor service (e.g. burst strength). However, the behavior of ATF cladding concepts must ultimately be tested using representative assembly geom-etries where broader system response under simulated LOCA conditions can be analyzed. A collaboration between Advanced Fuels Campaign (AFC) researchers and the Karlsruhe Institute of Technology (KIT) successfully used the unique capabilities of the QUENCH facility to subject a FeCrAl assembly to the time-temperature-pressure conditions anticipated of a LOCA and reflood to provide the first ever data on hydrogen generation and possible perturbations to core geom-etry (e.g. coolant channel blockage) for this ATF cladding material.

Project Description:

The QUENCH facility at KIT was established to answer critical questions regarding perturbations to the reactor core that may result from design basis accidents and postulated severe accidents. The design of the facility can accommodate test bundles containing numerous (typically 20-30) 2.5 meter long cladding segments temperature controlled using annular zirconia pellets heated by an internal tungsten element. The power provided to the heating elements coupled with extensive instrumentation allows for precise control of the simulated core conditions following a wide range of transients. During the test, temperature and flow conditions can be monitored in addition to hydrogen generation. Melting of core components can also be induced to examine redistribution and material interactions.

The objective of the first ever test of FeCrAl in QUENCH was a comparison of FeCrAl(Y) and ZIRLO under similar power and gas flow conditions. ORNL staff fabricated numerous FeCrAl(Y) (B136Y3) tubes and a larger shroud to the geometry required by KIT. FeCrAl spacer grids were also supplied to ensure that a representative ATF test was achieved. ZIRLO was tested previously by KIT researchers (QUENCH-15). The FeCrAl(Y) tubes were shipped to KIT and assembled to a representative assembly geometry. The QUENCH-19 test was then successfully executed on August 29, 2018. Initial data confirmed that the test was successful and online data collection active.

Accomplishments:

The primary driver for the present test was to establish the baseline performance of FeCrAl cladding with a specific emphasis on collection of data to benchmark MELCOR analysis. Although adaptation of material properties in MELCOR has been performed, to date no specific benchmarking activities have been executed due to a lack of experimental data. Test parameters were also matched to those of the previously completed QUENCH-15 test where ZIRLO was studied to enable a direct comparison of temperature and hydrogen generation.

Two tests were performed. In the first, a FeCrAl bundle was heated to 1000C, reached following roughly 4000 seconds. The test

was performed in an atmosphere of argon and superheated steam. At the end of this test a single rod was extracted from the bundle to facilitate subsequent measurement of oxide thickness and other properties as desired. In the second phase, the conditions described above were repeated and the 1000C temperature maintained for 2000 seconds. Power was then increased until the cladding temperature reached 1500C (Fig 1). Measurable hydrogen production began to occur once cladding temperatures exceeded 1400C. A liquid water reflood was then performed at 9100 seconds, indicated in Figure 1 by the rapid drop in the FeCrAl temperature and power reduction.

To date only preliminary analysis has been performed of the data collected during the test. The most notable outcomes are that no temperature excursion was observed during the high temperature test phase. Zirconium alloys such as ZIRLO will cause further heating of the core due to the exothermic oxidation reaction. This is shown in Figure 1,

where the ZIRLO temperature can be seen to increase rapidly beginning at approximately 6500 seconds. This in turn accelerates hydrogen production, also seen in Figure 1. This behavior is not observed during the FeCrAl test. Hydrogen production is thereby greatly reduced for the FeCrAl cladding. The reference ZIRLO test produced 48 g of hydrogen, while the FeCrAl test produced only 9.4 g despite being subjected to nearly 2000 seconds of additional time above 1200C.

Analysis of this data and examination of the microstructural evolutions of the cladding samples will proceed in FY19. Simulation of the QUENCH-19 test using MELCOR will also be performed to facilitate an improved understanding of the ability of existing accident performance codes to accurately predict the impact of non-zirconium cladding alloys on core behavior. These results will then be used to propose future QUENCH tests where conditions can be matched to those anticipated of FeCrAl during a LOCA for specific reactor designs.

ADVANCED REACTOR FUELS

3.1 FUELS DEVELOPMENT

Annular Extrusion Development

Principal Investigator: Randall Fielding
Collaborators: Brady Mackowiak, Ginger Dexter, Robert O'Brien

Annular fuel has been proposed as an advanced fuel concept, with initial results showing positive results. In addition to casting, extrusion has been shown to be a viable metallic fuel fabrication process. Extrusion will likely have advantages over casting for annular fuel fabrication because casting of thin walled tubes using standard casting technique has shown to be very difficult. This work looks into the feasibility of extruding annular U-Zr fuels.

Project Description:

EBR-II fuel was traditionally fabricated through casting of the fuel. Although casting has been proven to be an efficient process to fabricated large quantities of fuel it may not be as efficient for some advanced fuel concepts, such as annular fuel. A mechanically bonded annular fuel may be applicable to high burnup application, may have a positive impact on reactor design, and will not require sodium as a thermal transfer media, which may simplify the final disposition path for the used fuel. Because the fuel will be mechanically bonded a tighter control of surface finish and final dimensions as compared to traditional sodium bonded will likely be necessary.

In the metal forming industry extrusion is a common process that is used produce tubular products with good dimensional control and surface finish, although for a finer surface finish further warm or cold processing may be necessary. Another potential advantage of extrusion is the ability to co-extrude a composite product, producing a multi-layer product. Extrusion may be a feasible fabrication approach to produce annular fuel rods, with a fine surface finish, and tight dimensional control. If a suitable material can be extruded with the fuel, a fuel cladding chemical interaction barrier may also be applied at the same time while reducing contamination spread during the fabrication process. Another fuel concept that becomes feasible with extrusion is an axially graded fuel. In this work the feasibility of extrusion for advanced, especially annular, fuel fabrication is shown.

Accomplishments:

Dr. Robert O'Brien collaborated on the segmented billet experiment design, while Brady Mackowiak and Ginger Dexter were instrumental in extrusion processing and characterization of the extruded product.

A U-10Zr billet was cast and machined in the usual fashion. After machining the billet was cut into six segments. Three of the segments were cut using the lathe and mill which produced a non-oxidized surface with a surface finish of 16 - 32 μin, as determined by visual comparison with a standard. The remaining three segments were cut using electro-discharge machining, which leaves a darkened, or oxidized, rough surface finish. Figure 1 shows the segmented billet and zirconium can ready for assembly. Following assembly and seal welding, the billet assembly was heated to 800°C and extruded using a 6.05 mm die. Extrusion went well with no abnormalities noted in the process. After extrusion the extruded rod was sectioned for ease of handling during characterization. Radiography showed no internal voids or separation within the extruded product. Following radiography the billet was dimensionally characterized. Table 1 shows the overall length and diameter of the sections. Diameter measurements were taken at the approximate center of each section at two locations approximately 90° apart. Further microstructural characterization, in particular at the segment interfaces, will continue next year.

Figure 1. Segmented billet and zirconium can components ready for assembly.

Section ID	Length (mm)	Center Diameter (mm- 0°)	Center Diameter (mm- 90°)
1	255.57	5.96	5.96
2	258.75	5.93	5.92
3	257.18	5.91	5.89
4	255.57	5.87	5.91
5	257.18	5.77	5.75
6	258.75	5.80	5.71
7	258.75	5.72	5.83
8	258.75	5.63	5.70
9	257.18	5.69	5.79
10	255.57	5.62	5.71
11	257.18	5.63	5.70
12	174.63	5.56	5.46

Table 1. Length and diameter of the extruded segmented U-10Zr rod.

Following the success of a segmented billet the extrusion of an annular U-10Zr tube was performed. As with the previous extrusions, a standard U-10Zr billet was extruded and machined. Annular extrusion billets incorporate a central hole through the billet. During the extrusion process a mandrel, which is connected to the extrusion stem, is pass through the billet and is long enough to also pass through the extrusion die. As the material is extruded it passes over this mandrel forming the central hole. A vendor supplied extrusion tooling stack up schematic is presented in Figure 2. The billet assembly was heated to 800°C and the extrusion performed using an 8 mm die and 6 mm mandrel, dimensions are approximate. During the extrusion process no anomalies in the extrusion force were noted, however, after the extrusion was finished it was observed that the mandrel had broken during the extrusion process. Despite the broken mandrel a section of annular U-10Zr tubing was produced. Figure 3 shows the cross section of the annular product. The reason for the mandrel breakage is still under investigation. Forces on the extrusion press were well within historical amounts seen for other extrusions. This shows that extrusion of an annular product is feasible, although some tooling optimization is necessary. In addition to tooling optimization to produce longer annular sections, follow on work will include extrusion of zirconium clad products, and reduction of the zirconium thickness allow it to be used as a fuel clad chemical interaction barrier, and characterization of the cast products.
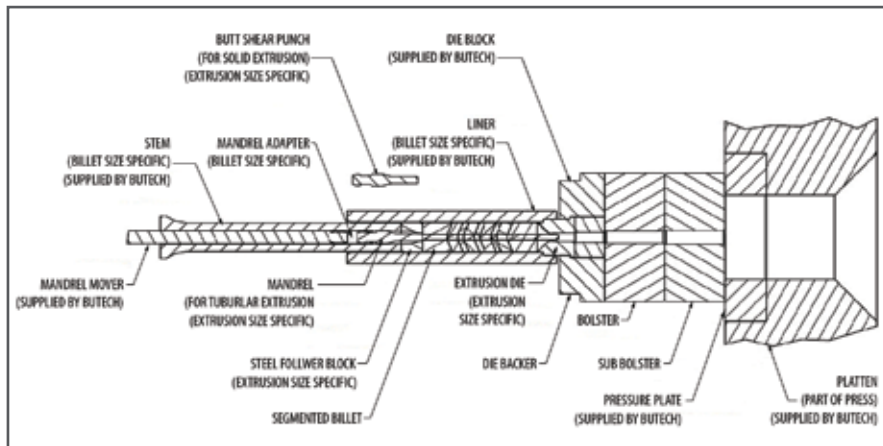
Figure 2. Extrusion press vendor tooling stack-up schematic. Note: Some of the components shown were not used in this work.

Figure 3. Annular U-10Zr produced through extrusion.

Alloy Optimization Casting and Characterization Studies

Principal Investigator: Michael T. Benson
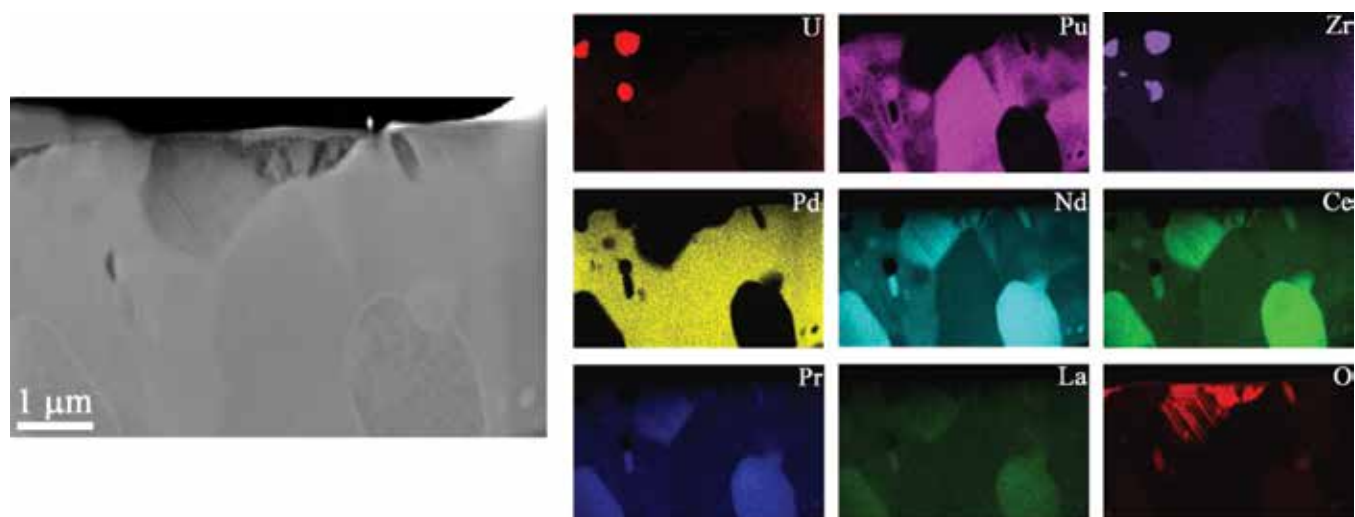*Collaborators: James A. King*

Figure 1. TEM EDS maps for a precipitate in U-20Pu-10Zr-3.86Pd-4.3Ln

Fuel-cladding chemical interaction (FCCI) occurs when the nuclear fuel or fission products react with the cladding material. A major cause of FCCI in U-Zr and U-Pu-Zr fuels during irradiation is fission product lanthanides (Ln), which migrate to the fuel periphery, coming in contact with the cladding. The result of this interaction is degradation of the cladding, and will eventually lead to rupture of the fuel assembly. Tin, palladium, and to a lesser extent antimony, are being investigated as minor component additives to control FCCI in metallic fuels specifically due to lanthanides. The role of the additive is to prevent FCCI by forming very stable intermetallic compounds with the lanthanides, thus preventing interaction with the cladding. Studies are underway to characterize the effects of these additives in a metallic fuel.

Project Description:

The technical objectives of this research are to investigate additives to metallic fuels for improved FCCI performance. Previous work on palladium has shown promising results for controlling FCCI. The current work is a continuation of that work with palladium, as well as expanding the investigation to include tin. The objective of this work is to characterize the microstructure of the metallic fuels with these additives, and to evaluate performance in out-of-pile tests. This work was carried out using U-Zr based fuels as well as U-Pu-Zr based fuels.

An additive that effectively controls FCCI will help the DOE meet its objectives of a safe, reliable, and economic reactor by significantly improving fuel performance. By preventing FCCI due to the fission

product lanthanides, cladding ruptures will be prevented, improving fuel safety and reliability, and higher fuel burn-up will be possible, thus improving reactor economics by decreasing the amount of fuel required, and decreasing the amount of nuclear waste generated.

Accomplishments:

The technical goals of this research is to explore the effectiveness of adding minor additives to a metallic fuel to bind fission product lanthanides, with the end goal of preventing or decreasing FCCI. To this end, both tin and palladium were investigated during FY18. The publications listed below detail the work performed to accomplish this goal.

The microstructures of U-10Zr-4.3Sn and U-10Zr-4.3Sn-4.7Ln are reported, as well as the diffusion couple results between U-10Zr-4.3Sn and lanthanides. In this diffusion couple test, the lanthanides aggressively attacked the Sn-Zr compounds in U-10Zr-4.3Sn, forming Sn-lanthanide compounds and releasing Zr back into the fuel matrix. In addition to using tin as a stand-alone additive, there was some experimental data suggesting tin and antimony may react differently towards the different lanthanides. Cerium and neodymium are not the same chemically, and do not always react the same, thus additives that target a specific lanthanide may be beneficial. To explore this, the microstructure of U-10Zr-2Sn-2Sb and U-10Zr-2Sn-2Sb-4Ln were

analyzed. Diffusion couples between these alloys and either Fe or the lanthanides are underway. It was not possible to determine from the microstructures alone if there is any difference in reactivity between tin, antimony, and specific lanthanides. The diffusion couples should provide the needed information.

To explore tin as an additive in a transmutation fuel, U-20Pu-10Zr-4Sn and U-20Pu-10Zr-4Sn-4.3Ln have been fabricated. The as-cast microstructure has been analyzed, and annealing the samples is underway.

The annealed microstructures of U-12Zr-4Pd and U-12Zr-4Pd-5Ln were investigated, as well as the as-cast structures for U-20Pu-10Zr-3.86Pd and U-20Pu-10Zr-3.86Pd-4.3Ln. Diffusion couples between the two Pu alloys and Fe have been run. The data is being analyzed. The annealed microstructure of the Pu alloys is also being analyzed. Figure 1 shows the Transmission Electron Microscopy (TEM) Energy Dispersive X-Ray Spectroscopy (EDS) map for a precipitate in annealed U-20Pu-10Zr-3.86Pd-4.3Ln. With Pu in the alloy, the only Ln-Pd compound is the 1:1 LnPd. The Ln7Pd3 compound, observed in U-12Zr-4Pd-5Ln, does not form. The result of this is too much lanthanides not being bound by Pd. Free lanthanides are present in the precipitates. A manuscript is in progress for both the annealed structures and the diffusion couples.

Fuel-cladding chemical interaction due to fission product lanthanides can be prevented using additives to react with the lanthanides, thus extending the lifetime and increasing the potential burn-up of a metallic fuel.

Fabrication of Rodlets for the Integrated Recycling Test

Principal Investigator: Randall Fielding
Collaborators: Blair Grover

> **The Advanced Fuel Campaign with its partner programs have fabricated and are currently irradiation testing metallic fuel produced through recycling of used oxide fuel.**

The Integrated Recycling Test (IRT) was a collaborative test with domestic and international partners which leveraged the Advanced Fuels Campaign Outboard A (AFC-OA) drop in test design, as well as experiment assembly equipment. The test was designed to include fuel samples produced from U-TRU feedstock which had been recovered from spent fuel recycling. The test was fabricated, assembled and inserted into the Advanced Test Reactor (ATR).

Project Description:

The disposition of used nuclear fuel has been an issue for many years for the nuclear industry. Several countries have proposed the concept of a closed fuel cycle in which Light Water Reactor (LWR) fuel is recycled and re-used in a fast reactor in order to fission or transmute actinide materials, which reduced the burden on a geologic repository. Although this concept has been put forward, the feasibility has not been experimentally tested. The IRT is an initial step towards confirming the behavior of recycled fuel.

Used mix-oxide (MOX) fuel was recycled to produce a metallic uranium-transuranic feedstock. This feedstock was used to produce 2 fuel compositions, which included two separate levels of simulated fission products. The fuel samples were encapsulated into the standard AFC-OA drop in tests and inserted into the Advanced Test Reactor. This tests marks the first time as oxide fuel was recycled into a metallic feedstock used to produce a fast reactor type fuel. The success of this test will show the feasibility of a closed fuel cycle as a method of reducing the volumetric and time burden a geologic repository would face.

Accomplishments:

The goal of the IRT test was to produce an irradiation test using feedstock recovered from recycled used oxide fuel. MOX fuel was recycled to produce a metallic U-TRU feedstock button. This button was used to produce two fuel compositions in which the level of simulated fission products was varied. A master alloy of high enriched uranium,

zirconium, and rare earth elements was added to the U-TRU feedstock to produce the final fuels. Table 1 shows the final composition of the fuel. It should be noted that the original nominal fuel composition included 1% and 3% of the rare earth simulated fission products, however, after casting an average of 14.5% of the rare earth elements were lost. The nominal simulated rare earth ratio was 53Nd:25Ce:16Pr:6La, however, as seen in Table 1 the ratio changed to an average ratio of 41Nd:39Ce:15Pr:6La. Reported relative error on the rare earth elements was 5-10%, this may account for some of the ratio change. The change in ratio also indicates a preferential crucible interaction or oxidation of some of the rare earth elements. Once the fuel was cast and

cut to length it was encapsulated in either HT-9 or FC92 steel cladding. Of the five rodlets, one had a chromium fuel cladding chemical interaction barrier layer and another had a TiN barrier layer. Table 2 shows the test matrix, and Figure 1 shows the completed rodlets. After final inspections the rodlets were encapsulated in containment capsules. Figure 2 shows a typical example of the final capsule, ready for shipment to the reactor. The containment capsule provides a safety barrier between the sodium and fuel and the reactor coolant in case of a cladding breach, but also provides thermal separation allowing the peak inner clad temperatures to reach prototypic liquid metal cooled fast reactor temperatures. The cladding temperature is dependent

Nominal Comp	Total U	Total TRU	Zr	Total RE	Nd	Ce	Pr	La	Other*
U-24TRU-10Zr-1RE	65.76	23.31	9.6	0.86	0.34	0.34	0.13	0.05	0.98
U-24TRU-10Zr-3RE	64.66	24.00	9.25	2.54	1.07	0.97	0.37	0.12	

Table 1. Summary of final fuel compositions in wt%.

Rodlet ID	Nominal Comp	Cladding	Coating	Capsule Fill Gas	Cladding Temp
IRT1-R1	U-24TRU-10Zr-1RE	HT-9	NONE	64.5He:35.5Ar	550±25°C
IRT1-R2	U-24TRU-10Zr-3RE	FC92	NONE	66.2He:33.8Ar	525±25°C
IRT1-R3	U-24TRU-10Zr-3RE	HT-9	TiN	85.1He:14.9Ar	600±25°C
IRT1-R4	U-24TRU-10Zr-3RE	FC92	Cr	60.0He:40.0Ar	500±25°C
IRT1-R5	U-24TRU-10Zr-1RE	FC92	NONE	65.6He:34.4Ar	525±25°C

Table 2. Summary of the test IRT-1 test matrix.

on the size of the gap between the rodlet and capsule and the composition of the gas in that gap. Fill gases can be varied from 100% helium for a higher conductivity gas gap to 10% helium in 90% argon if lower conductivity gas is required. As built calculations using the final as-built dimensions and fuel compositions indicated that two of the five capsules would run excessively hot in the reactor, leading to likely failure. In order to mitigate this risk a swaging process was developed to reduce the size of the rodlet-capsule gas gap. The swaging process successfully reduced the capsule diameter while not impacting the rodlet on both capsules while maintain design tolerances. Based on the re-worked dimensions calculations showed that one capsule,

IRT1-C3, would still run too hot. This capsule was disassembled using standard machining techniques and the rodlet recovered. Based on as built dimensions and fuel compositions a new, higher conductivity, fill gas was specified. Table 2 shows the fill gas compositions of all capsules. It was originally planned to insert all five capsule in reactor cycle 164A (June 2018), however, because IRT1-C3 had to be re-built it was inserted a cycle later in 164B (September 2018). In conclusion, five AFC-OA drop in style irradiation tests were fabricated using U-TRU feedstock recovered from recycled used MOX fuel.

Figure 1. Completed IRT-1 rodlets ready for encapsulation.

Figure 2. Typical IRT-1 capsule ready for shipment to the reactor.

Updated Advanced Reactor Fuels Handbook

Principal Investigator: Dawn E. Janney

Figure1. Caption

Figure 2. Caption

Transmutation of minor actinides such as Np, Am, and Cm in spent nuclear fuel is of international interest because of its potential for reducing the long term health and safety hazards caused by the radioactivity of the spent fuel. One important approach to transmutation involves incorporating minor actinides into U-Pu-Zr alloys, which may also include rare-earth elements (La, Ce, Pr, Nd) as a result of previous reprocessing as part of a closed fuel cycle.

It is important to understand the properties of U-Pu-Zr alloys, including those that also include minor actinides (Np, Am) and rare-earth elements (La, Ce, Pr, and Nd). The existing experimental data is widely scattered, and much of it was published before ~1975. The Metallic Fuels Handbook critically reviews the available experimentally based knowledge of several key properties of U-Pu-Zr alloys with and without minor actinides and rare-earth fission products.

Project Description:

Transmutation of minor actinides such as Np, Am, and Cm in spent nuclear fuel is important because of its potential for reducing long-term health and safety hazards caused by the radioactivity of the spent fuel. One important approach to transmutation (currently being pursued by the DOE Fuel Cycle Research & Development Advanced Fuels Campaign) involves

incorporating the minor actinides into U-Pu-Zr alloys, which can be used as fuel in fast reactors. These fuels are well suited for electrolytic refining, which leads to incorporation rare-earth fission products such as La, Ce, Pr, and Nd. It is, therefore, important to understand not only the properties of U-Pu-Zr alloys but also those of U-Pu-Zr alloys that include minor actinides (Np, Am) and rare-earth elements (La, Ce, Pr, and Nd) in concentrations relevant for transmutation fuels.

Previous revisions of this Handbook included information about elements and alloys in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system. The primary emphasis of this revision is increasing the detail and comprehensiveness of information on U-Zr and U-Pu-Zr alloys, including information about a number of properties that are new to the Handbook. In addition, this revision updates information about the entire U-Pu-Zr-Np-Am-La-Ce-Pr-Nd system from earlier revisions.

The handbook is intended to serve several audiences, including project leadership and management, researchers, and modelers.

Accomplishments:

The vast majority of the research and writing for this revision of the Handbook was done by the P.I., who is an INL employee. Some information was obtained from

previous INL publications whose authors are Douglas E. Burkes, James I. Cole, Randall S. Fielding, Steven M. Frank, Thomas Hartmann, Timothy A. Hyde, Dennis D. Keiser, Jr., J.Rory Kennedy, Andrew. Maddison, Robert D. Mariani, Scott C. Middlemas, Thomas P. O'Holleran, Cynthia A. Papesch, Bulent H. Sencer, and Leah N. Squires. Although some of these people are no longer employed at INL, all were INL employees when they performed the work used in the Handbook. Personnel from the INL Research Library provided invaluable help in finding obscure references, Dr. Pavel Medvedev (an INL employee) assisted with Russian-language translation, and Dr. Steven Hayes (an INL employee) provided insight and valuable guidance.

It is expected that the Metallic Fuels Handbook will be a multi-year project with new versions available periodically. Although the titles of revisions varied, all were given the INL publication number INL/EXT-15-36520. Each revision updated the data in the previous versions, as well as expanding the scope of the Handbook. The original Handbook (in 2015) included information about elements and alloys in the U-Pu-Zr system. Revision 1 (in 2016) contained information about the elements, binary alloys, and ternary alloys in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system. Revision 2 (in 2017)

added information about alloys whose nominal compositions include at least four elements. Revision 3 (in 2018) featured in-depth coverage of experimental results from U-Zr and U-Pu-Zr alloys. Each of the revisions includes information about phases and phase diagrams, heat capacities, thermal expansion, and thermal conductivities and diffusivities for the materials it covers; Revision 3 also includes some information about mechanical properties of some alloys. The Handbook has a strong bias in favor of experimental data when available, but supplements the experimental data by modeling results as needed. The Handbook also contains numerous references to other modeling papers as a guide for researchers.

The 2017 revision was split into two documents because of its size. Part 1 contained information about U-Zr, Pu-Zr, U-Pu, and U-Pu-Zr alloys with and without minor actinides (Np, Am, Cm), rare-earth elements (La, Ce, Pr, Nd, Gd) and Y. Part 2 contained information about elements and other alloys in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system. This organization has been retained for the 2018 revision, which is almost 400 pages long.

Because of the importance of U alloys with 10% Zr, experimental results from this alloy were published as a separate journal paper in August, 2018.

The transmutation fuel package has supplied fuel samples for thermal and microstructural characterization as well as the fuel and assembly of the AFC-3F irradiation test.

The handbook provides a single resource for data and critical reviews of important properties in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system based on an extensive survey of widely scattered, often obscure, sources.

Production of Americium and Neptunium Metal

Principal Investigator: Leah Squires
Collaborator: James King, Randall Fielding

> ***Currently, pure americium and neptunium metals are in very short supply across the DOE complex and a reliable method for isolating the pure metals from available oxide, mixed metals and mixed oxide materials is necessary.***

Pure americium and neptunium metals are crucial components in the fabrication of transmutation fuels. Unfortunately, neither neptunium nor americium is available in pure metal form. However, a number of oxides, mixed metals and mixed oxides that include americium and neptunium are available. Methods have been devised to separate and/or reduce the materials available in order to obtain pure neptunium and americium metals.

Project Description:

The disposal of spent nuclear fuel is a challenge currently facing the nuclear power industry due to the hazards associated with long term storage of the material once it is removed from the reactor. Key to the reduction of these hazards is the reduction or elimination of minor actinides with long half-lives including neptunium and americium that are present in spent fuel. Transmutation is one method to potentially resolve this issue. Transmutation aims to incorporate the long lived actinides into new fuel that can be placed in a fast reactor where the elements of concern will fission into products which have shorter half-lives and can be more easily stored in disposal facilities. In order to better understand transmutation and its potential, it is necessary to fabricate small quantities of fuel with minor actinide additives and perform thorough characterization and irradiation testing. In addition, the thermal characterization of pure americium and neptunium metals

is incomplete and the pure metal is needed to perform basic thermal measurements. Currently, there is very little pure americium or neptunium metal in existence; and therefore, it is necessary to develop methods to isolate these metals from the oxide, mixed metal and mixed oxide feedstock material that is available.

Accomplishments:

A new approach to isolate of both americium and neptunium metals was investigated over the course of FY18. In the past, americium metal was isolated from a mixed metal starting material consisting of approximately 90% americium and 10% neptunium. The difference in vapor pressures of the two metals was used to perform a distillation separation and approximately 12g of americium metal was successfully obtained using this process. Unfortunately, this specific starting material was exhausted at the end of FY17; therefore, a new mixed metal starting material containing approximately 10% americium and 90% plutonium was used for distillation at the beginning of FY18. Early distillation experiments using the method previously applied to the americium/neptunium mix showed that the material could not be efficiently collected using the existing set-up. This was mostly due to size limitations. The small crucible diameter did not provide the surface area necessary to vaporize significant amounts of americium from such a small quantity of starting material. The material that did vaporize

Figure 1. Neptunium oxide starting material.

Figure 2. The same material arc melted under 4% hydrogen cover gas and cooled for 15 minutes under 4% hydrogen.

Figure 3. The same material arc melted under 4% hydrogen cover gas a second time and cooled under argon.
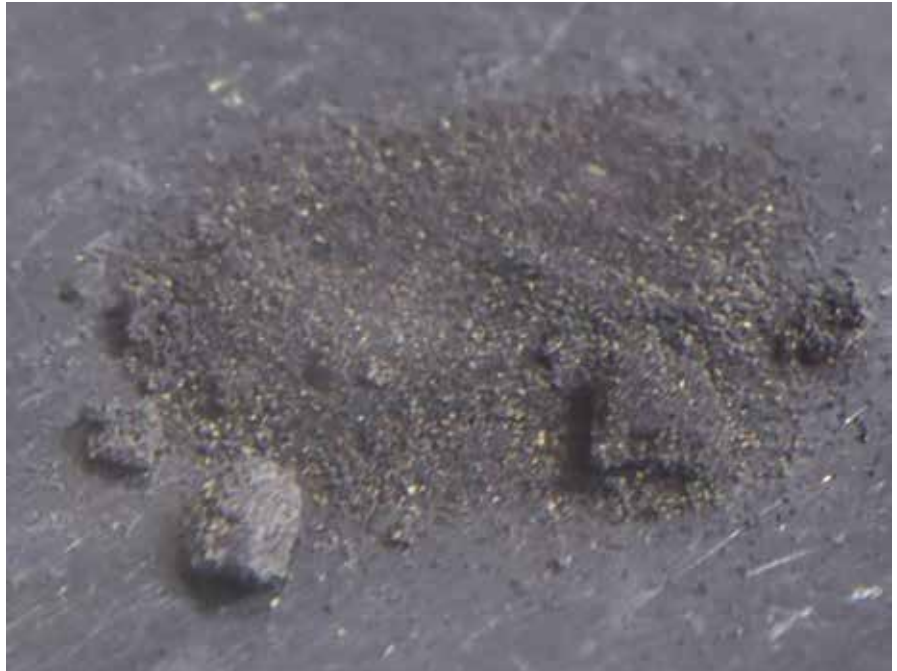

Figure 4. Americium oxide starting material.

deposited in a thin layer such that the retrieval of the americium without tantalum contamination from the crucible walls proved to be impossible. Neptunium metal was produced in previous years via a direct oxide reduction using calcium metal as a reducing agent and calcium chloride salt. One issue that arose for both the neptunium and americium metals was storage of the material as it was found to oxidize readily even sealed in an airtight container and kept in an argon atmosphere glovebox. A method was devised early in FY18 to go directly from the oxides of both americium and neptunium to the metals using an arc melter to reach very high temperature under a hydrogen containing cover gas. The idea was that the extreme temperature would force a separation of the oxygen from the americium or neptunium and the hydrogen in the cover gas would collect the released oxygen and carry it away as water vapor. The method was tested using zirconia in a non-radiological facility with great success. Offsite analysis showed the material to be 99% metallic. The method was then applied to both americium oxide and neptunium oxide in the argon atmosphere glovebox with varying percentages of hydrogen in the argon cover gas. The material underwent a dramatic visual change, but analysis results are still pending. If this technique is further refined it could provide a method to produce the metal at the time it is needed, thereby eliminating the storage requirement and the difficulty of protecting the material from oxidation over long periods of time. It would also be much faster than current methods of direct oxide reduction for neptunium isolation or distillation for americium isolation.

Figure 5. The same material after arc melting under 15% hydrogen cover gas.

3.2 CLADDING DEVELOPMENT

Improvement of Mechanical Properties of HT-9 Steels by Modified Thermomechanical Treatment

Principal Investigator: T.S. Byun
Collaborators: J.P. Choi, T.G. Lach – PNNL, S.A. Maloy - LANL

The sodium-cooled fast reactors have been designed for operation in a wide temperature range of 300–550°C and ongoing development efforts for a higher thermal efficiency require improved performance of core materials at higher temperatures. Thus far, the 12Cr-1MoVW (HT-9) steels with fully-tempered martensitic structures have been used as core structure materials for the fast reactors not only because they exhibit high resistance to irradiation-induced embrittlement, thermal and irradiation creep, and void swelling but also because they show little compatibility problem with liquid sodium coolant; however, the irradiation-induced embrittlement in low-temperature irradiation, void swelling at very high doses (> 150 dpa), and high-temperature strength still limit expansion of the capability of HT-9 steels to higher doses and temperatures required in advanced fast reactors. In this work package, therefore, significantly improved high-temperature mechanical performance with the HT-9 alloy has been pursued by developing new thermomechanical processing routes.

Project Description:

This research aims to develop new processing routes for HT-9 steels that can deliver improved high-temperature mechanical properties required in advanced fast reactors. For the fiscal year, a series of new processing routes were designed and the samples after those processes were tested and compared in search of an optimized microstructure that can demonstrate both high strength and high fracture toughness in the reactor operation temperature range. Application of 16 different thermomechanical treatments (TMTs) to two HT-9 steels was performed considering the guidance from the comprehensive thermodynamics simulation performed earlier in the project. Finer microstructures usually demonstrate higher resistance to radiation damage as well as better mechanical performance. Among the numerous processing routes that have been explored in the steel development, only a limited number of processing methods, including the rapid cooling that can effectively refine the quenched lath structure, are feasible for processing

> *Improved high-temperature mechanical properties were obtained through the new processing route (WQ-500 or 600°C) that combines a rapid quenching and a low degree of tempering.*

thin core components such as fuel cladding and fuel duct. In this effort, therefore, we planned to produce a quenched and tempered martensitic structure with ultrafine laths and precipitates by combining the rapid quenching and nontraditional tempering treatments. We planned to carry out basic characterization for the newly processed materials, including the basic Scanning Electron Microscopy (SEM) and Transmission Electron Microscopy (TEM) for selected TMTs and uniaxial tensile testing over 22–600°C range for all TMT conditions. Finally, the fracture resistance (J-R) testing in the temperature range of 22–600°C was carried out for selected TMT conditions to use the data as a primary selection criterion for improved processing. Comparing the test results

should enable to propose a few non-traditional TMTs after which the HT-9 steels demonstrate significantly improved mechanical properties. It is expected the main outcomes of this research will help produce high performance cladding samples for more in-depth characterization.

Accomplishments:

Design of thermomechanical processes for HT-9 steels and testing and evaluation of the processed samples were performed to determine new processing conditions that can produce optimized microstructures of HT-9 steels with improved mechanical properties. In designing thermomechanical processes it was considered that the proposed treatments, such as the rapid cooling and following tempering treatment, should be feasible for the practical

processing of thin core components such as fuel cladding and duct. It was also considered that high toughness might be achieved only when the selected processing route can avoid formation of too much or coarse brittle phases. We were able to produce tempered martensitic structures with ultrafine laths and precipitates by a combination of rapid quenching, i.e., water quenching(WQ), and relatively low degree of (or nontraditional) tempering. Sixteen TMTs were designed and applied to two HT-9 steels, heat-3 and heat-4, in search of such new processing routes that can yield exceedingly high strength and high toughness in the operation temperature range of 300–550°C. Summarized below are the key observations in the mechanical testing and relevant conclusions.

Figure 1 presents plots for the temperature dependence of yield stress and fracture toughness in various processing conditions. A wide range of strengths were measured depending on the degree of tempering, in particular. It is notable that both HT-9 alloys before tempering or after single tempering below 600°C can achieve ultrahigh yield stresses above the one GPa mark. Both the yield strength and the tensile strength of HT-9 steels monotonically decreased with test temperature, regardless of different alloy compositions and TMT routes; the decreasing of those strengths was slow up to 500°C

and became more rapid between 500°C and 600°C. Overall, the final tempering temperature turned out to be the single most important factor controlling the strength of HT-9 steels. Meanwhile, the ductility parameters showed rather complex behavior as their temperature dependence depended on processing route, and thus on the strength of the materials. In general, the rankings of room temperature (RT) strength parameters were approximately reversed in the ductility parameters.

The fracture toughness at RT is generally quite high (i.e., > ~200 MPa√m) except for the samples after no or low-temperature (< 500°C) tempering and the temperature dependence of fracture toughness above RT is strongly dependent on the degree of tempering. It is notable that the two processes with the HT-9 heat-3, WQ with 500°C tempering and WQ with 600°C tempering, yielded increased fracture toughness at 600°C: 250 and 230 MPa√m, respectively, which can be positively compared to the typical K-range of 150–200 MPa√m for the typical HT-9 steels. In summary, the mechanical properties of HT-9 steels varied widely with their processing routes, particularly on the degree of tempering. Some tailored TMTs, e.g., combination of a rapid quenching and a limited tempering (WQ-500°C for instance) yielded excellent strength and improved fracture toughness.
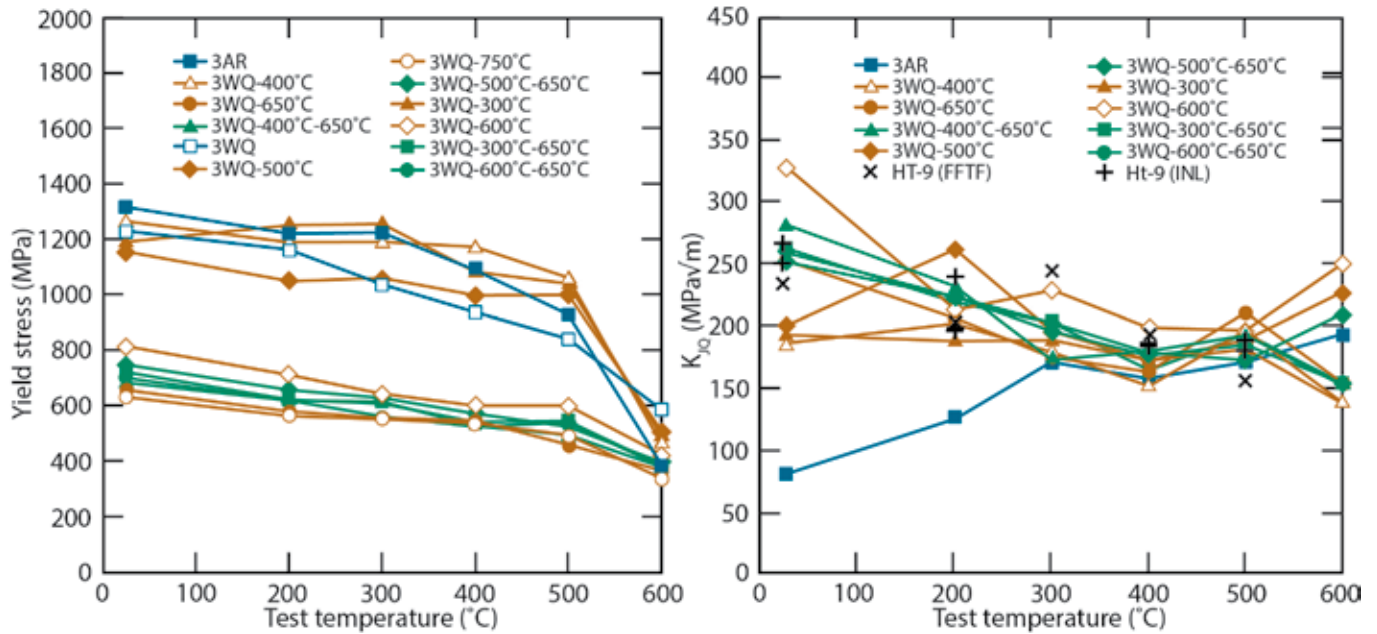
Figure 1. Comparison of yield stress (left) and fracture toughness (right) for the HT-9 steel heat-3 after various heat treatments (note: AR = as-rolled; WQ = water quenched; WQ-300°C (for example) = water quenched and tempered at 300°C; FFTF = fact flux test facility).

Evaluation of the Use of Ion Irradiation to Simulate Neutron Radiation Effects in MA957

Principal Investigator: Jing Wang, PNNL
Collaborators: Mychailo B. Toloczko, PNNL

There is an ongoing need to evaluate the effects of neutron irradiation on the structure and mechanical properties of existing and candidate nuclear reactor core structural materials. However, performing neutron irradiations is expensive, time consuming, creates radioactive material, and the available neutron irradiation facilities across the world have become much more limited. Using ion irradiations to estimate neutron irradiation effects has been a known technique for more than 40 years. Ion irradiations are quicker to perform, much less expensive, and do not produce radioactive material. But until recently it was not a favored technique because there was a concern that ion irradiations do not sufficiently estimate neutron irradiation effects. In recent years, there has been a strong push to improve ion irradiation techniques with the intent of providing a better simulation of neutron irradiation effects, and studies are underway to evaluate the effectiveness of using ion irradiations to simulate neutron irradiation effects. The results of one such study are presented here.

The use of Ion irradiations to estimate how materials behave in a nuclear reactor core environment may allow for much more rapid development of advanced materials for reactor core structural components and fuel assemblies.

Project Description:
The work described here is a comparison of the microstructure (structure at the micrometer to nanometer scale that plays a significant role in the properties of structural materials) of the oxide dispersion strengthened ferritic steel MA957 (a candidate advanced material for nuclear reactor fuel cladding) after exposure to neutron irradiation and ion irradiation. It is part of a larger effort at PNNL to investigate, optimize, and use ion irradiations to estimate neutron irradiation effects to develop improved alloys for core structural materials in nuclear reactors. Being able to use ion irradiations as a first line irradiation effects screening tool allows for much more rapid and cost efficient development of advanced alloys. It has the potential to benefit all reactor types including existing light water reactors and future advanced nuclear reactors such as sodium cooled reactors and molten salt reactors. The development of such materials is very important because it makes for a more robust nuclear reactor that will operate more cost effectively.

Accomplishments:
The approach to evaluating the effectiveness of ion irradiations to simulate neutron irradiation effects on reactor core structural materials has been to compare to the microstructure of materials after neutron and after ion irradiation. This is a common approach that has been used several times over the last 40 years. An important difference with the work at PNNL compared to previous research is that the ion irradiations were conducted using optimized methods to simulate neutron irradiation effects. A key part of PNNL's contribution to the AFC program over the last eight years has been the optimization of ion irradiation techniques.

There are many candidate materials in existence, and a thorough evaluation of the effectiveness of ion irradiations requires performing comparative studies on a wide range of candidate materials. The results presented here on just one alloy (MA957) are thus understood to be a step in a larger, ongoing effort. The neutron irradiations were conducted in the DOE-NE owned Fast Flux Test Facility (FFTF) in the late 1980's while the ion irradiations were conducted using Cr ions at the Kharkov Institute of Physics and Technology (KIPT) in the Ukraine. Irradiation temperature and dose were similar between the two irradiation methods.

Many qualitative similarities in the microstructure of neutron and ion irradiated MA957 were observed after moderate amounts of irradiation exposure. Some examples are 1) radiation induced alpha-prime precipitate formation occurred over the same irradiation temperature range, 2) low amounts of void swelling were observed, and 3) there was good stability of the oxide particle population that is present in this material. However perhaps not surprisingly, quantitative assessment of the microstructure revealed some differences. For instance the alpha-prime population in the ion irradiated material had a different

size and number density than in the neutron irradiated material as shown in Figure 1. Comparative studies just getting started on other materials are showing similar results with many qualitative similarities but quantitative differences.

The outlook thus far is that optimized ion irradiation techniques are showing promise as a first line screening tool that will provide insight into the neutron irradiation response of a material, perhaps with the ability to allow weeding out materials with clear performance issues. Efforts to fully understand the value will continue for the next several years.
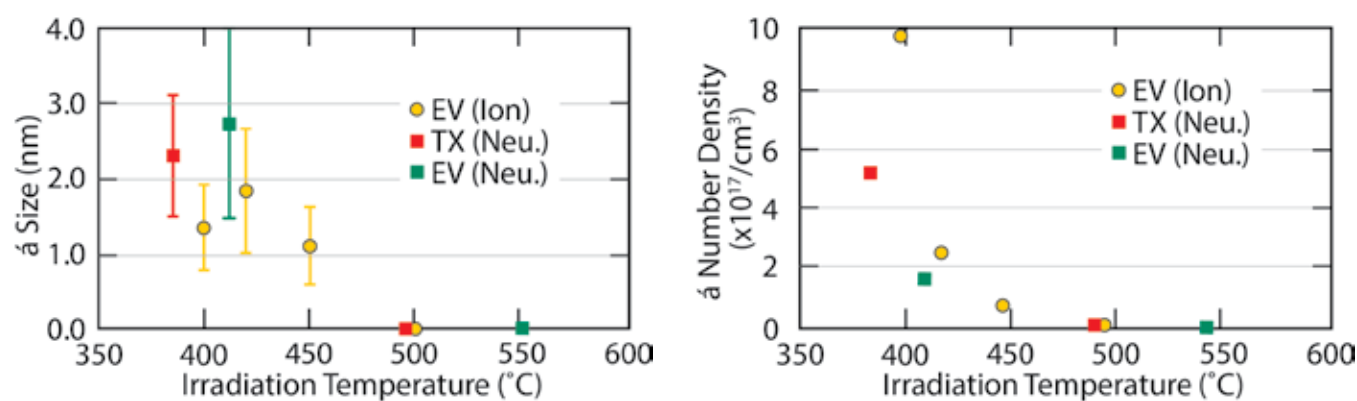

Figure 1. Mean size (radius) and number density of alpha-prime precipitates as a function of irradiation temperature for MA957 irradiated by neutrons or ions.

Fate of Injected Ions During Ion Irradiation of Core Structural Materials

Principal Investigator: Mychailo B. Toloczko, PNNL
Collaborators: Jing Wang, PNNL

> **Novel material analysis techniques are being used to understand the details of ion irradiation conditions allowing ion irradiations to be more effectively applied to estimating neutron irradiation effects in reactor core structural materials.**

There is an ongoing need to evaluate the effects of neutron irradiation on the structure and mechanical properties of existing and candidate nuclear reactor core structural materials. However, performing neutron irradiations is expensive, time consuming, creates radioactive material, and the available neutron irradiation facilities across the world have become much more limited. Using ion irradiations to estimate neutron irradiation effects has been a known technique for more than 40 years. Ion irradiations are quicker to perform, much less expensive, and do not produce radioactive material. But until recently it was not a favored technique because there was a concern that ion irradiations do not sufficiently estimate neutron irradiation effects. In recent years, there has been a strong push to understand and improve ion irradiation techniques with the intent of providing a better simulation of neutron irradiation effects. The results of one such study to better understand ion irradiations are presented here.

Project Description:

When a material is bombarded with neutrons, the majority of the neutrons readily leave the material after collisions with atoms within the material. However during ion irradiations, nearly all of the ions remain in the material, and for high dose ion irradiations that are needed to assess the performance of core structural materials over the lifetime of the reactor or clad and duct during high burn up conditions, a large number of ions are deposited. The depth distribution of where these atoms initially come to rest is readily determined, and this distribution has a strong peak. Microstructural examinations of ion irradiated materials are typically conducted at regions away from this peak without regard for the possibility that the injected atoms may diffuse away from their original stopping point, potentially affecting microstructural evolution in the investigated region. For example these atoms could redistribute to grain boundaries, and it would be unclear if the enhancement of an element on a grain boundary was due to the inherent behavior of the material or due to the injected atoms. Thus it is important to understand the behavior of self-ions in order to fully assess the ability for ion irradiations to simulate neutron irradiations.

Understanding the fate of injected atoms is especially difficult when the ion being used is also a primary constituent element of the material being studied. This use of "self-ions" is actually strongly preferred because there is less chance that the injected atom will strongly affect the microstructure. While it possible to track the combined concentration of the injected self-ions and the base element using traditional composition measurement tools such as energy dispersive x-ray spectroscopy (EDS) in a transmission electron microscope, the actual fate of the injected atoms cannot be tracked by this method. Researchers at Pacific Northwest National Laboratory (PNNL) have devised a way to track these injected atoms by ion irradiating a material using self-ions of low natural abundance isotopes of an element. Because of their low natural abundance, these injected atoms are readily distinguished from the constituent element in the material using mass sensitive composition measurement tools such as secondary ion mass spectroscopy (SIMS) and atom probe tomography (APT). Initial results if self-ion fate studies are presented here.
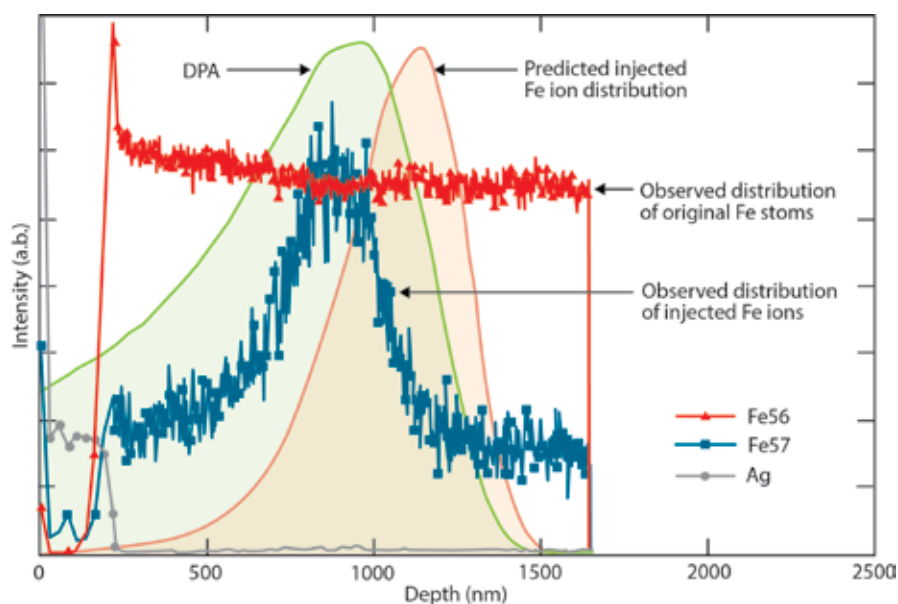
Figure 1. SIMS measurement of iron distributions overlaid on predicted irradiation dose and injected atom profiles. Data are scaled and offset individually along the y axis for visualization purposes. As a result, the relative strength of the signal among different elements does not correlate with relative atomic abundance.

Accomplishments:

The focus for this initial study was to observe the fate of injected self-ions during ion irradiation of HT-9 that is a reference clad and duct material for sodium cooled reactors. The primary element in HT-9 is iron, so the cathode from which the ions are extracted during ion irradiation was made from Fe57. In natural iron, Fe57 has an abundance of only ~2%, so injected Fe57 atoms are readily distinguished from the iron originally present in HT-9. The ion irradiation was performed at a sodium cooled reactor relevant temperature of 400°C. Because irradiation is known to greatly enhance diffusion, there was a strong expectation that there would be substantial redistribution. However, as shown in Figure 1, the observed width of the injected atom distribution (as determined by SIMS) approximately matches that of the predicted width in absence of diffusion. The measured distribution is however offset from the predicted distribution, but rather than this being due to diffusion, it is likely due to inaccuracies in the calibration data used by the algorithm that predicts the injected atom distribution. Other aspects of possible redistribution such as diffusion down grain boundaries has yet to be investigated, but this initial observation of a lack of any gross redistribution due to radiation enhanced diffusion at reactor relevant irradiation temperatures is excellent news for Fe self-ion irradiation of ferritic alloys because it appears that microstructural investigations can be performed in regions nearly free of injected atoms.

Fast Reactor Cladding Material Development

Principal Investigator: Stuart Maloy
Collaborators: Eda Aydogan (LANL), Ben Eftink (LANL), Dave Hoelzer (ORNL), G. Robert Odette (UCSB), John Lewandowski (CWRU), J. DeCarlan (CEA), D. Sornin (CEA)

This research is critical to the application of advanced Oxide Dispersion Strengthened (ODS) steels to engineering applications as it addresses one of the most difficult tasks which is to produce tubing from these radiation tolerant, high strength steels and the stability of these steels at very high doses.

The Nuclear Technology Research and Development (R&D) program is investigating options to transmute minor actinides. To achieve this goal, new fuels and cladding materials must be developed and tested to high burnup levels (e.g. >20%) requiring cladding to withstand very high doses (greater than 200 dpa) while in contact with the coolant and the fuel.

Project Description:

To develop and qualify materials to a total fluence greater than 200 dpa requires development and testing of advanced alloys and irradiations in fast reactors. Specimens of previously irradiated HT9 specimens are being irradiated in a fast reactor to high doses (>200 dpa). In addition, improvements in the radiation tolerance of HT-9 are being made through minor changes in the composition. Advanced radiation tolerant materials with fine oxide dispersion strengthening are also being developed to enable the desired extreme fuel burnup levels. This fine microstructure provides an alloy with high strength at high temperatures and excellent radiation tolerance (e.g. reduced void swelling and ductility retention at low temperatures) but also increases the difficulty of producing engineering parts (e.g. thin walled tubes) from these advanced materials. Thus, in this project, research is underway to produce tubes using techniques such as high temperature hydrostatic extrusion, intermediate temperature plug drawing and low temperature pilger processing.

Accomplishments:

Progress has been made at understanding the effects of small levels of nitrogen on the properties of ferritic/martensitic steels. Heats of Fe12Cr0.2C alloys were produced with low (10 wppm) and high (500 wppm) levels of nitrogen. Ion irradiations were performed on heats with and without nitrogen to doses of 0.5 and 1.0 dpa at 300C using the Ion Beam Materials Laboratory at Los Alamos National Laboratory (LANL). The results showed a much higher uniform density of loops in the high nitrogen alloy (Figure 1) where it is postulated that the nitrogen is nucleating loops under irradiation. Higher dose irradiations were performed using iron ions to a maximum dose of 23 dpa showing that hardening saturates after 2-3 dpa.

Research continues on a large heat (50 kg) of a nanostructured ferritic alloy (14YWT) that was produced through a collaborative
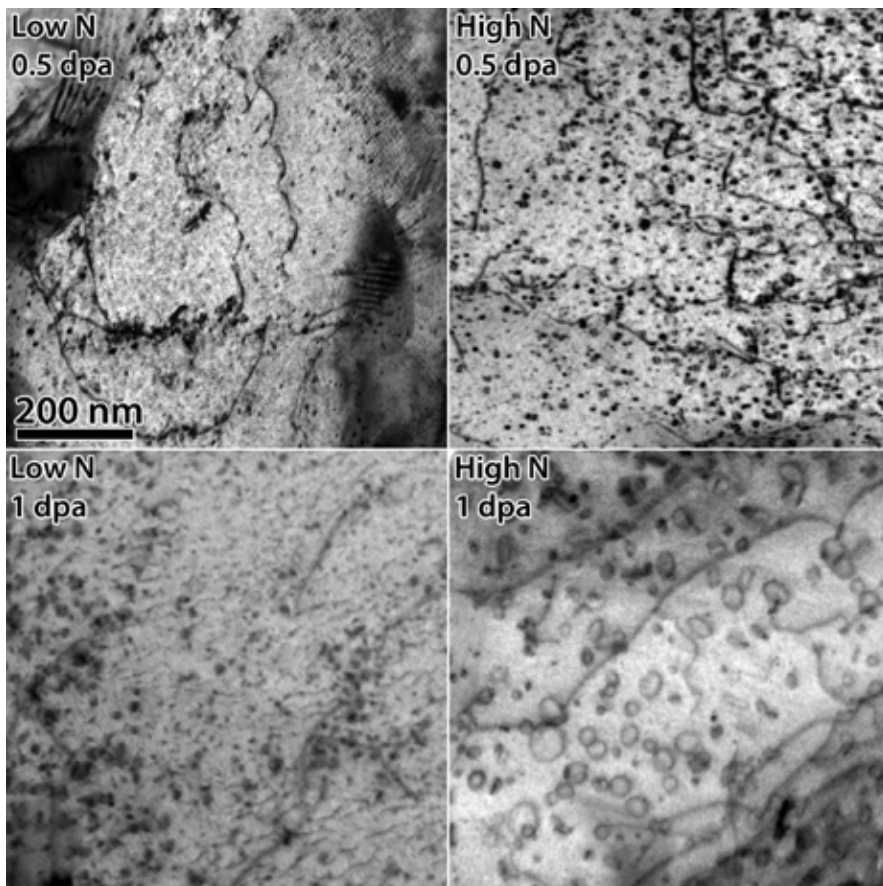
Figure 1. Transmission Electron Microscopy (TEM) micrographs of low and high nitrogen Fe-12Cr-0.2C alloys after ion irradiation to 0.5 and 1.0 dpa using protons.

effort between LANL, University of California Santa Barbara (UCSB) and Oak Ridge National Laboratory (ORNL) named FCRD-NFA1. Significant progress has been made at processing this alloy into tube form. Through a bilateral collaboration with Centrale Energie Atomique (CEA), pilger processing was used to produce a 915 mm long tube with a 10.73 mm OD and wall thickness of 0.5 mm (Figure 2). In addition, tubing was produced at Case Western Reserve University

(CWRU) using hydrostatic extrusion at 815C (Figure 3). Tubes were produced from 14YWT material supplied by CEA as well as from DOE material. Final dimensions of the tubes were 230 mm long, 7 mm diameter and a 0.5 mm wall thickness. Local texture in the tubes was characterized using scanning electron microscopy electron backscatter detection (SEM-EBSD) to show a large grain size in the pilger processed material but finer grain size in the hydrostatically extruded material (Figure 4). Future work will center on producing tubes for future fueled irradiations in the Advanced Test Reactor (ATR).

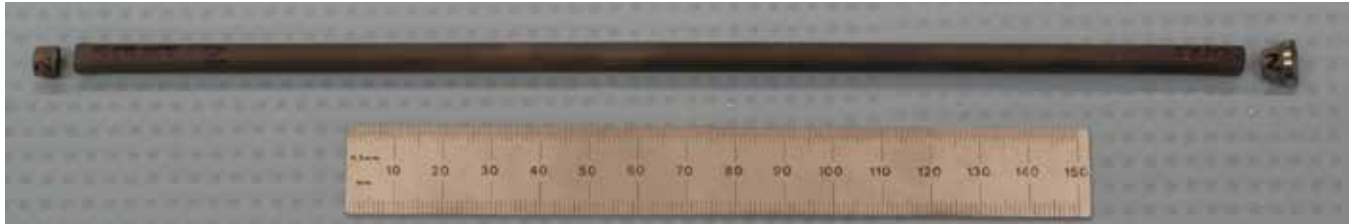*Figure 2. Image showing 14YWT tube after extrusion to 915 mm in length using pilger processing at CEA.*
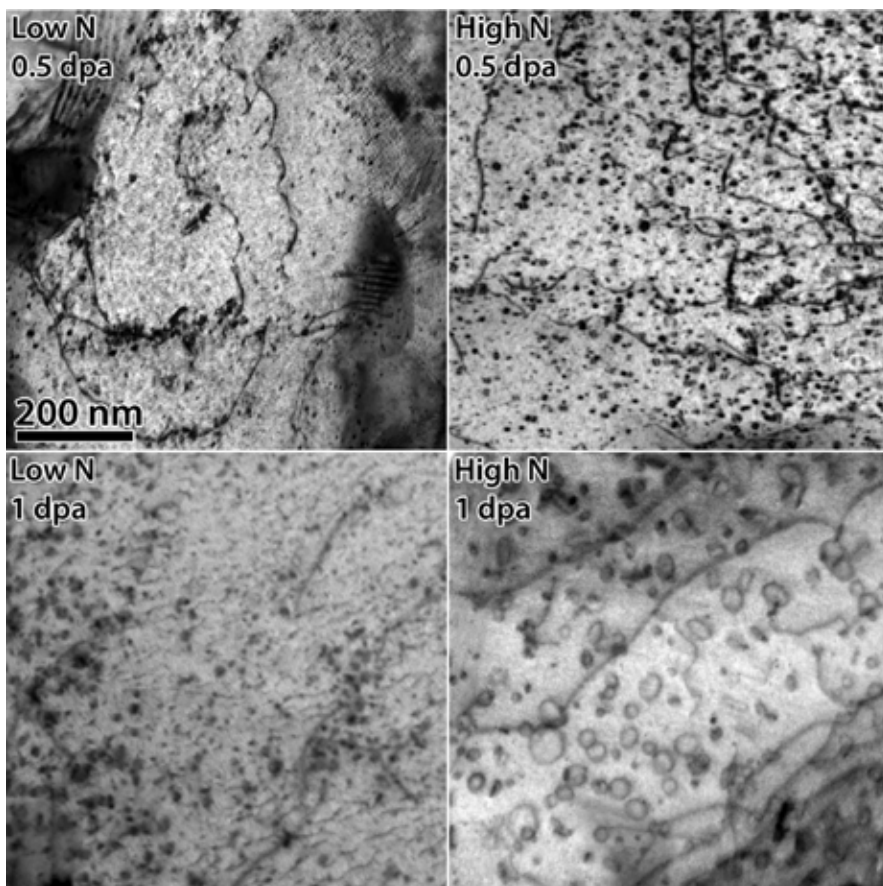
Figure 4. SEM-EBSD images showing the variation in grain size and texture on 14YWT tubes produced by hydrostatic extrusion (HE) (on left) and pilger processing (on right).

Postirradiation Examination on EBR-II Legacy Metallic Fuel with Minor Actinides

Principal Investigator: Jason M. Harp
Collaborators: Luca Capriotti

Figure 1. Neutron radiography of the fuel pin X501-G591.

Research into enabling technologies that allow for the destruction of long lived minor actinide isotopes generated in irradiated nuclear fuel through transmutation in fast reactors is a long term goal of the Advanced Fuels Campaign (AFC). In an effort to better understand the fuel performance implications of adding minor actinides to a fuel system, the several AFC experiments have examined the performance of metallic, nitride, and ceramic fuels in approximated fast reactor neutron spectra at the Idaho National Laboratory (INL) Advanced Test Reactor (ATR). In addition and complementary to ATR testing, systematic characterization of experiments irradiated in true fast reactors are performed in order to assess ATR in qualifying fast reactor fuel. For this purpose, postirradiation examination are performed on a selected fuel pin from the EBR-II

X501 experiment. This experiment contained unique U-19Pu-10Zr metallic fuel with the addition of minor actinides (Am). This is the only irradiated EBR-II experiment fabricated with minor actinides at the start of irradiation.

Project Description:
The X501 experiment was meant to characterize the behavior in pile of minor actinides bearing metallic fuel. Two sister pins (X501-G582 & X501-G591) were irradiated in EBR-II. Postirradiation Examination (PIE) were limited and only performed on X501-G582. A new systematic PIE campaign on X501-G591 has been performed at the Hot Fuel Examination Facilities (HFEF) of the Material and Fuel Complex (MFC) at INL. The PIE campaign consists of non-destructive and destructive examination such as, visual inspection, neutron radiography, dimensional exams, gamma spectroscopy, optical

microscopy, and chemical analysis to evaluate burnup and transmutation. To date, the behavior of the fuel pin looks unaffected by the addition of minor actinides compared to the expected behavior of conventional U-19Pu-10Zr ternary metal fuels.

Accomplishments:

Baseline postirradiation examination on X501-G591, a unique metallic fuel with addition of minor actinides irradiated in EBR-II, has been performed at INL. The average pin
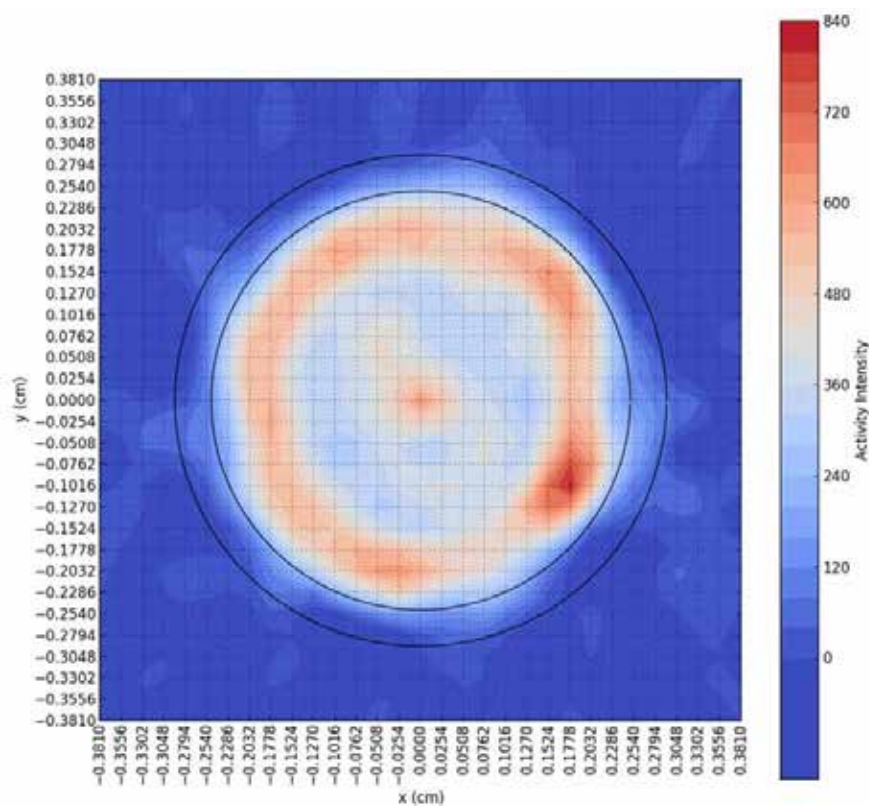

Figure 2. Cs-137 2D redistribution at the middle of the fuel slug. Tomography reconstruction.

burnup was 7.6 atomic % heavy metal. Neutron radiography of the examined pin (see Figure 1) revealed typical metallic fuel behavior for this burnup. The fuel appears to have swelled radially to fill the cladding, the axial swelling of 1.8% is in line with expectations, and a small amount of fuel is dissolved into the sodium at the top of the fuel column. Gamma spectrometry also revealed prototypic behavior. In addition to axial gamma spectrometry scans, tomographic gamma spectrometry scans were performed that revealed the Cs-137 distribution at mid-plane. This is shown in Figure 2, and reveals that some of the Cs is located in the interior of the fuel while the majority is located on the periphery near the inner cladding. The black lines in Figure 2 represent the location of the cladding in the tomographic scans. Cesium is often observed to travel dissolved in the sodium. The Cs signal from the center of the fuel pin may be an indication of sodium occupying some of the porosity in the fuel. This is sometimes known as sodium logging and is a poorly understood but significant performance effect in metallic fuel. Further research into this sort of Cs signal is planned to better understand if this technique can further illuminate the sodium logging phenomenon. Optical microscopy taken from the center of the fuel stack at the same location as the gamma tomography is shown in Figure 3. The location of large porosity in this cross section generally corresponds to the location of Cs-137 in the gamma tomography. Constituent redistribution at this location is not as pronounced as often shown in literature for U-Pu-Zr metallic fuel, but the central region does show evidence of Zr migrating towards the center of the fuel and a phase change. This can be seen in the central area of Figure 3 stretching out in a radius of about 250µm from the center of the fuel cross section. These exams generally show good performance for this alloy and that there is little to no detrimental effects from the addition of americium to this fuel. However, electron microscopy is needed to characterize this fuel system more fully at the micro scale and determine the location of Am in the fuel. This is planned for next year.

Characterization of EBR-II legacy fuel pins is of paramount importance in order to assess the applicability of using ATR testing to evaluate transmutation fuel performance.

Figure 3. X501-G591 metallography near the axial mid-plane of the fuel slug

Observations of Irradiated Annular and Low Smear Density Metallic Fuels

Principal Investigator: Jason M. Harp
Collaborators: Luca Capriotti, Fabiola Cappia

The AFC-3C, AFC-3D, and AFC-4A capsule irradiations were irradiated at the Idaho National Laboratory (INL) Advanced Test Reactor (ATR) and underwent postirradiaton examination (PIE) recently. These irradiations were planned to test several different fast reactor fuels that could be used to facilitate ultra-high burnup applications in sodium fast reactors. Several different alloys, fuel geometries, bonding materials, and the use of additives were tested in ferritic-martensitic HT-9 cladding.

Project Description:

This research facilitates the development of fuel technologies for different missions that can be used in a sodium fast reactor or other similar advanced reactor concepts. One of the technologies tested is He bonding of metallic fuel to cladding. If this technology can be successfully demonstrated it could be used to possibly eliminate sodium bonding of the fuel to the cladding. This is an important enabling technology that simplifies the final disposal of fuel used in a "once-through fuel cycle" advanced reactor. These tests also investigated the irradiation effect of adding lanthanide gettering additives. This technology may enable high burnup fuel operation beyond historically established limits by preventing fuel cladding chemical interaction embrittlement of the cladding. The final major technology investigated is the effectiveness
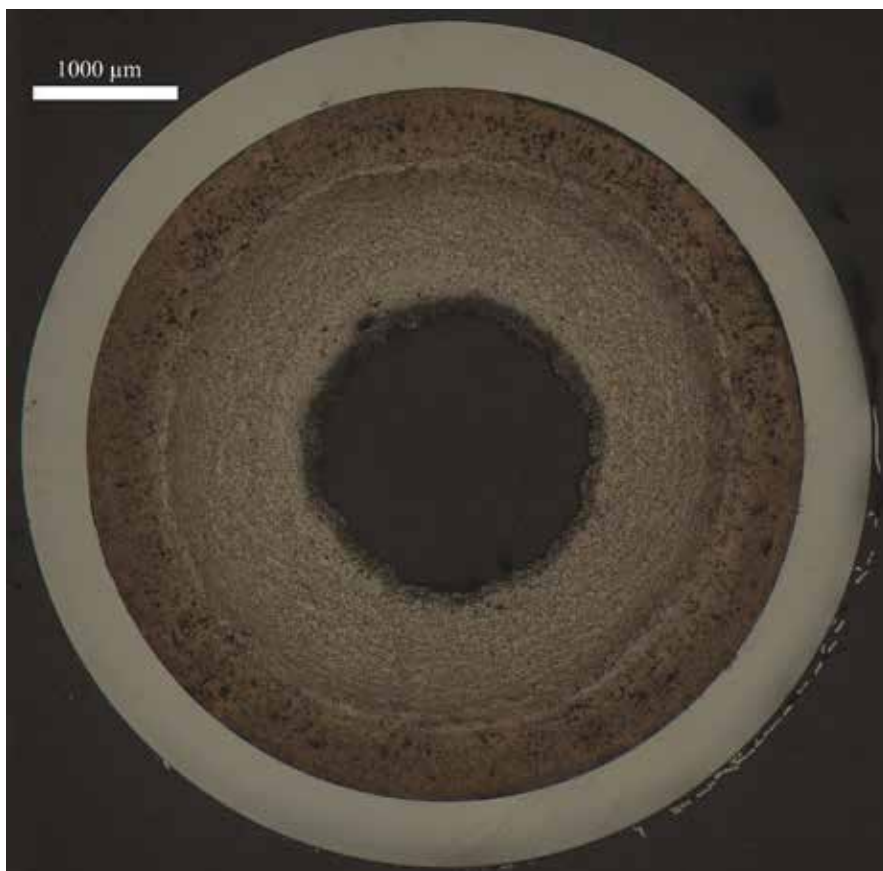

Figure 1. Transverse cross section of AFC-3D R1 (U-10Zr, 55% smear density, annular, He bonded, 4.3 atom % burnup). This fuel has performed well to moderate burnup and shows no sign of fuel cladding chemical interaction which demonstrates the need to carefully control the dimensions of fuel when using helium bonding.

of utilizing alternative alloys that simplify the behavior of the fuel under irradiation. These alloys may be easier to model using advanced modeling and simulations tools. Additionally testing alternative alloys helps to highlight different physical phenomena in the fuel creating further opportunities to refine and enhance modeling tools.

Accomplishments:

The AFC-3C, AFC-3D, and AFC-4A experiment took various different U-Mo, U-Zr, U-Pd-Zr, U-Mo-Ti-Zr, and U-Mo-Ti-Zr-Pd alloys to burnups of 2.1% to 4.5% FIMA (fissions per initial heavy metal atoms) and fission densities of 6.13×1020 to 1.27×1021 fissions / cm³. These alloys were evaluated against the

Figure 2. Transverse cross section of AFC-3D R5 (U-4Pd-13Zr, 55% smear density, annular, He bonded, 2.9 atom % burnup). This fuel has performed well with high additive loading which demonstrates the need to balance additive loading with excess Zr to ensure good early life fuel performance.

> *These postirradiation exams have highlighted important technological data necessary to enable once though sodium fast reactor fuel that do not have sodium bonded fuel.*

historical fuel performance of previously irradiated fuel from literature. Fuel performance was irradiated through a suite of postirradiation examination techniques including neutron radiography, gamma spectrometry, dimensional inspection, fission gas release measurements, chemical burnup analysis, and optical microscopy. The examinations took place at the Idaho National Laboratory, Materials and Fuels Complex (MFC), Hot Fuel Examination Facility (HFEF) and Analytical Laboratory. The irradiations were performed at relatively aggressive temperatures that resulted in significant interaction between the cladding and the fuel in many cases. In spite of this, there were no in-pile cladding breaches in these tests. The U-Zr alloys performed better than the U-Mo or U-Mo-Ti-Zr alloys. Some of the fuel-cladding interaction issues seen in helium bonded annular fuel that was not machined to tight tolerances appear to have been resolved by machining the outer diameter of annular fuel in this irradiation. Higher Zr concentrations in additive bearing fuel appears to have improved the performance of Pd additive fuel.

Fuel Safety Testing – TREAT (AR)

Principal Investigator: Dan Wachs
Collaborators: Todd Pavey

The Transient Reactor Test (TREAT) Fuel Safety Testing for Advanced Reactors has continued with efforts in developing fuel safety research plan for advanced metal fuel alloy experiments, design separate effects tests to study metal fuel behavior, and advanced reactor irradiation capability designs, specifically working on Temperature Heat-sink Overpower Response module (THOR) and a metals fuel loop design.

Project Description:

The objective of the TREAT Fuel Safety Testing for Advanced Reactors is to develop and qualify metal fuels with improved performance, reliability, and safety characteristics during normal operations, accident conditions and with reduced waste generation. The goal of fuel development for advanced fast reactors is to create a nuclear fuel system that performs at high burnup while supporting the use of recycled actinides. Nuclear Science and Technology (NS&T) transient testing program has worked on research and development of a metals fuel test plan that can be used in the following years as a guidance document for future metal fuels transient experiments to help improve and qualify metal fuels. The transient testing program developed metal fuels separate effects tests in TREAT. The separate effects tests designs concentrated on irradiation impacted phase diagram and annular fuel stability under overpower conditions. The transient testing program is working on conceptual design of THOR for separate effects testing in the future. The MK III sodium test loop that was used in past transient experiments prior to 1994 is being redesigned for future metals fuels transient testing. The NS&T transient testing program is working on a conceptual design of a metal fuels test loop.

Accomplishments:

Metallic fuel is a mature technology with a sufficient performance database to support licensing technology within the bounds of the existing database. The primary objective of this program is to reconnect to historical knowledgebase, maintain, and develop the technical basis for metallic fuel performance in off-normal conditions. These objectives may also provide crosscutting Research and Development (R&D) capabilities and expertise to support R&D needs of oxide fuel of interest to many in the international community.

A presentation titled, "New Generation of Transient Testing for SFR Fuels at TREAT," was presented at the ANS meeting in June 2018 by Colby Jensen.

Completed development of a strategic plan for future Fuel Safety Experiments. This was written by Colby Jensen and Dan Wachs with assistance from Hen Ban from the University of Pittsburgh. This document outlines a strategic plan for safety R&D for metallic fuels. The specific objectives of the safety research program for metallic fuels are to:

- Perform and support investigations of performance limiting technical issues for reactor designs (i.e. Versatile Test Reactor (VTR), commercial designs, etc.),

- Provide experimental evaluation of fuel behavior under power-to-cooling mismatch scenarios for the development of advanced fuel designs and behavioral models,

- Identify fuel failure mechanisms and important failure thresholds while providing various levels of experimental characterization of these to develop, improve, and validate modeling predictive capability,

- Provide confirmatory and qualification experimental evaluations for fuel designs to establish margins and support validation of fuel and systems modeling tools.

> *Fuel safety research is a vital part of developing and qualifying metal fuels with improved performance, reliability, and safety characteristics.*

The objectives of this document are to:

- Provide a guiding framework for the fuel safety research program supporting development and maturation (and ultimately the licensing basis) of metallic fuels for the Nuclear Technology Research & Development (NTRD) program,

- Be a mechanism for consensus and communication for DOE programs with specific interest in the performance and behavior of metallic fuels,

- Be updated annually to reflect evolving requirements and needs.

Conceptual design of THOR for separate effects testing for Sodium Fast Reactor (SFR) fuels has progressed by Nicolas Woolstenhulme. Some progress was also made in pre-conceptual design of the Metals Fuel Loop development.

Design of the Integrated Recycle Test

Principal Investigator: Gary Povirk
Collaborators: Dan Chapman, Doug Dempsey, Randall Fielding, Nate Oldham, Emily Swain

> *Among other objectives, the Integrated Recycle Test has experiments comprised of a number of different barrier materials that, if successful, have the potential to dramatically increase the burnup and high temperature capabilities of fast reactor fuel systems.*

The Idaho National Laboratory (INL) is working with an international partner to develop and test metallic nuclear fuel systems that will enable a closed fuel cycle that is both cost-effective and proliferation. In terms of in-reactor fuel performance, a key objective is to design the fuel system to such that it can achieve high burnup, which in turn will reduce the amount of reprocessing of fuel required per unit of energy produced. A challenge in this regard is to reduce the attack of the cladding by lanthanide fission products; in these experiments different barrier materials were placed between the fuel and the cladding so that the structural integrity of the cladding will be maintained.

Project Description

Fuel rods for advanced fast-spectrum reactors are typically irradiated within cylindrical capsules; a narrow gas-filled gap between the rodlet and capsule serves to raise fuel and cladding temperatures from levels in the Advanced Test Reactor (ATR)

coolant to values that are prototypical for a sodium-cooled reactor. A challenge in this project was that the irradiation times were constrained to be relatively short. Given the abbreviated irradiation times and that the objective of the experiments was to assess the ability of different barrier materials to protect the cladding, there was a strong desire to accelerate diffusion processes by running the tests at relatively high temperatures. However, the small gas-filled gap between the rodlet and the capsule and the fabrication tolerances of these components lead to high uncertainties in the temperatures that will actually develop in-reactor. The task at hand for the team was to provide best-estimate power levels for every fuel rod for the duration of the experiments and to choose gas compositions (mixtures of helium and argon) that would strike a balance between the desire to accelerate the interdiffusion of constituents and the perceived risk of running the test at too high a temperature given the uncertainties that are inherent in the design.

Accomplishments

The first step in the design of the experiment were scoping studies that involved physics calculations (Emily Swain), thermal analyses (Dan Chapman) and fuel fabrication (Randy Fielding) to provide rough estimates of the fuel enrichments required to reach desired burnup rates and temperatures. Once fuel rods were fabricated, a key to this project was to perform analyses based on as-built fuel rod properties that provided best-estimates of heat generation rates (Emily Swain) and temperatures (Dan Chapman) throughout the entire course of the experiment for a variety of gas compositions. These calculations represent first-of-a-kind studies since as-built projections of in-reactor conditions are typically only performed on a cycle-by-cycle basis. The analyses provided a complete picture of the experiments as a function of time and gas-gap composition, which aided in the selection of gas compositions that provided the best balance between accelerating chemical interactions and the risk to the experiment by running a too high a temperature. Selection of gas composition in a timely manner was also essential for fabricating and inserting the experiment into the ATR on schedule. Figure 1 provides but one example of the numerous analyses that were performed in support of these tests.
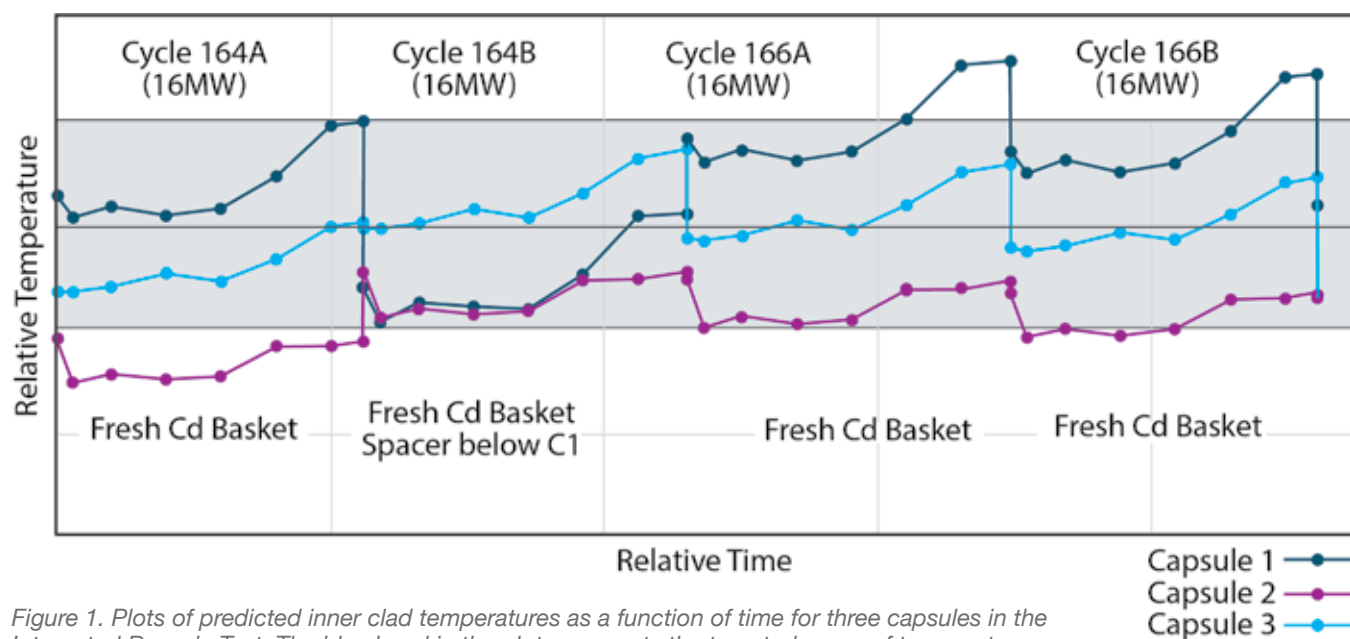

Figure 1. Plots of predicted inner clad temperatures as a function of time for three capsules in the Integrated Recycle Test. The blue band in the plot represents the targeted range of temperatures for the experiment.

3.4 FUEL PERFORMANCE MODELING

Modeling Porosity Migration in LWR and Fast Reactor MOX Fuel Using the Finite Element Method

Principal Investigator: Stephen Novascone
Collaborators: Pavel Medvedev, John W.Peterson, Yongfeng Zhang, Jason Hales

An engineering-scale finite element simulation of pore migration in oxide fuel has been developed. The porosity field is governed by an advection-diffusion equation which is coupled to the fuel temperature and stress fields through the thermal conductivity and volumetric heat source term. The engineering-scale porosity equation models the microscopic process of vapor transport of fuel across pores, taking into account thermal and vapor pressure gradients within the fuel. In the simulations, the porosity is initialized to a constant value at every point in the domain, and as the temperature gradient is increased by application of a heat source, the pores move up the thermal gradient and accumulate at the center of the fuel in a time frame that is consistent with experimental observations. Results from representative simulations are provided to demonstrate the new capability, and we show that a sufficiently high power ramp rate limits restructuring and leads to a corresponding increase in fuel temperature. multidimensional results.

Project Description:
After start-up and attainment of full power and nominal operating temperature, ceramic, or mixed oxide fuel (UO_2 and MOX) experiences micro- and engineering-scale restructuring, which has a profound effect on the bulk properties of the fuel. Restructuring mostly affects MOX fuel in fast reactors due to the higher temperatures, but can be relevant to light water reactors if temperatures go outside of normal operating ranges, such as during a loss of coolant accident. As such, the objective of this study is to consider oxide fuel in general, using models from a variety of sources. Here, restructuring means that the fuel pellet develops distinct regions from fuel center to fuel surface. Going from the center of the fuel outward in sequence, these regions are characterized by a central void, a region of increased density, and then a region with radially aligned grains and un-restructured grains. These regions, which depend on both the temperature value and temperature gradient, form radial isosurfaces within the fuel. The salient feature

of restructuring is the movement of pores. Under large thermal gradients, pores introduced during fabrication are generally greater than 1 μm in size, and are uniformly distributed throughout the pellet. Pores move toward the center of the pellet where they eventually coalesce into a large central void. Speculation regarding the manner in which the central void forms has led to the concept of a vapor transport mechanism in which fuel at the surface of the hot side of a pore vaporizes, then condenses on the cool side of the pore. The result is a net flux of pores toward the pellet center. The emphasis in the present work is on the application of these models to engineering-scale finite element simulations.

Accomplishments:

Central void formation in oxide fuel is known to be a strong function of rod linear power. As shown in Figure 1a, the diameter of the central void increases with an increase of linear heat generation (LHGR). To investigate response of the newly implemented pore migration model in BISON, a series of calculations of radial porosity distribution were performed

(a) Micrographs of void formation at 19.7 kW/m (left), 31.1 kW/m (middle), and 46.0 kW/m (right) [23].

(b) BISON imulation results at different linear power showing the effect of LHGR on radial porosity distribution.
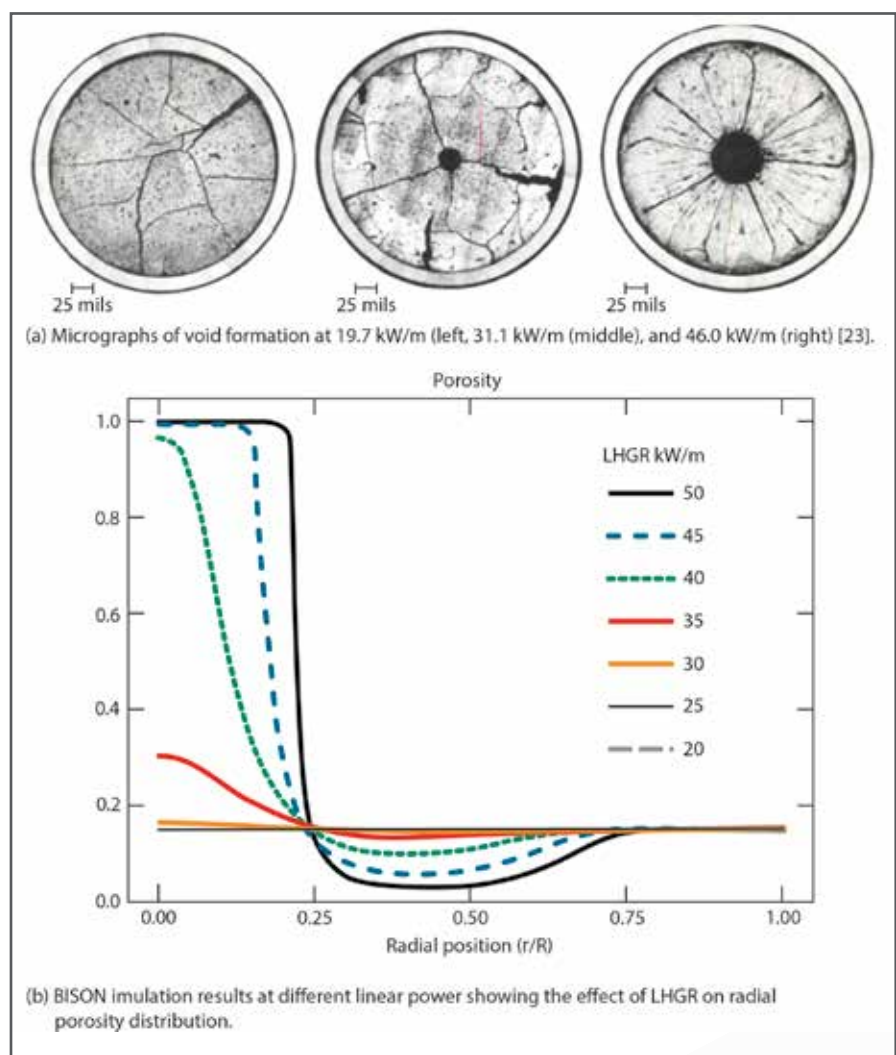
Figure 1. Top: micrographs of void formation at different linear powers. Bottom: BISON Calculation for the porosity field variable vs. normalized radial position from the 1D axially symmetric simulation of the fuel pellet at varying LHGRs. This shows the pore migration model adequately reproduces the trend of increasing void diameter with increasing power.

for LHGR of 20 to 50 kW/m in 5 kW/m increments. Results of these calculations are shown in Figure 2b. It was found that BISON has reproduced the trend of increasing central void with an increase of LHGR. Future work will be focused on extensive modeling of fast reactor oxide fuel irradiation experiments and comparing measured central void diameter with that calculated using BISON.

Results for the offset case are shown in Figure 2. The contour plot shows the peak temperature and porosity is offset from the geometric center, which is expected due to the offset fuel pellet within in the cladding. The offset position of the fuel pellet creates a gap that is larger on one side, resulting in correspondingly higher temperatures in the direction of the larger gap. Also note the elliptic shape of the central void with the major axis orthogonal to the thermal gradient. The offset and shape of this void is qualitatively similar to the micrograph shown in Figure 3.

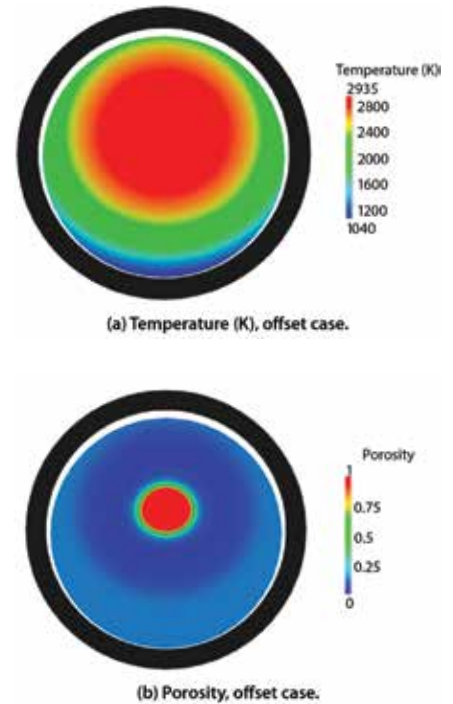(a) Temperature (K), offset case.

(b) Porosity, offset case.

Figure 2. 2D calculation showing temperature and porosity contours in a restructured fuel pellet in an offset (a, b) and position relative to the cladding. The as-fabricated porosity was 15%. Power was ramped from 0 to 37 kW/m in 104 seconds. Cladding exterior was held at 600 K. The pellet and cladding were thermally coupled using a meshless contact approach. For the offset case, the contour plot shows the peak temperature and porosity is offset from the geometric center, which is expected due to the offset fuel pellet within in the cladding. The offset position of the fuel pellet creates a gap that is larger on one side, resulting in correspondingly higher temperatures in the direction of the larger gap. Also note the elliptic shape of the central void with the major axis orthogonal to the thermal gradient. The offset and shape of this void is qualitatively similar to the micrograph shown in Figure 3.
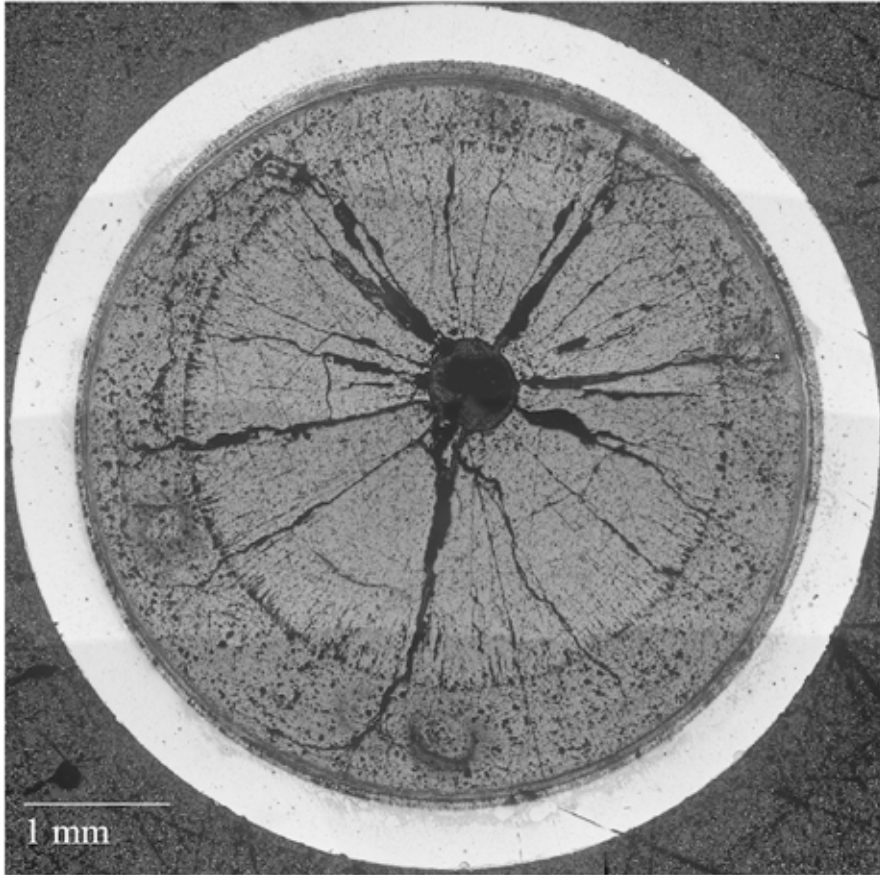
Figure 3. Micrograph of the U0.75Pu0.20Am0.03Np0.02O1.98 fuel irradiated at an average power of 321 W/cm. Compare the micrograph shown here to the porosity field calculation in Figure 2 for the offset case. Both figures show an elliptic-shaped central void offset from the geometric center.

Metallic Fuel Modeling

Principal Investigator: Cetin Unal (LANL) and Christopher Matthews (LANL)
Collaborators: Naveen Prakash (LANL), Jacob Hirschhorn (LANL, UF), Garrison Stevens (LANL), Blake Wilkerson (LANL)

Through the development, implementation, and calibration of mechanistic models can BISON is quickly becoming a useful tool for predicting metallic fuel behavior for future applications.

The BISON fuel performance code is well underway to being extended to model zirconium based metallic fuel. The challenges presented by the advanced U-Zr and U-Pu-Zr fuel types have resulted in a coupled thermo-mechanical-species diffusion simulation required to describe historical observations, as well as predict future performance. Advances in fundamental models such as swelling and zirconium diffusion will help provide the necessary foundation for advanced models such as fuel cracking and fuel-cladding chemical interaction (FCCI). Benchmark comparisons to past post irradiation examinations are utilized to assess model performance, with advanced calibration techniques playing a role when data is sparse or unavailable.

Project Description:

The objective of this research is to provide a code that can predict behavior of zirconium based metallic fuel for use in future advanced Gen IV commercial or test reactors such as the Versatile Test Reactor (VTR) currently under development. The favorable thermal conductivity and proven safety record of metallic fuel experienced during operation of the Experimental Breeder Reactor (EBR-II) lends U-Zr and U-Pu-Zr fuel naturally to the requirements of enhanced safety in next generation reactors. With fuel-cladding chemical interaction as the primary limiting factor of metallic fuels, enhancements in understanding and modeling of the key phenomena involved in FCCI can help bolster the economic viability of metallic fuel through extending the fuel lifetime. Simultaneously, the coupled nature of nuclear fuel requires the implementation of many fundamental models in order to provide a baseline capability for advanced concepts and off-normal behavior.

Due to the sparse availability of modern data, historical EBR-II data is used as a primary source of information for model formulation and comparison. Previous codes have been limited to heavily empirical models due limited resources, understanding, and data. By leveraging many of the capabilities within BISON, advanced models that capture complex behavior can be implemented in a viable way. In addition, on-going work within the Advanced Fuels Campaign (AFC) has helped bolster the limited data set with new irradiations and higher fidelity data collection. Modern computational tools such as Bayesian calibration have been implemented to help bridge over limited data. In general, metallic fuel modeling in BISON aims to create a tool that is predictive in order to provide confidence for core designers in the safe, reliable, and efficient use of metallic nuclear fuel.

Accomplishments:

During FY17, effort was focused on applying Bayesian calibration techniques to the zirconium redistribution model. This was completed utilizing the best available metallic fuel models in BISON and EBR-II data to assess the viability of the models. Failure to match the species redistribution profile across different datasets led us the conclusion that either the underlying BISON models are providing the wrong state variables, or the zirconium redistribution model is missing or masking physics relevant to species redistribution. Following

this, FY18 focused on improving the BISON simulations through robustness improvements and development of a new swelling model, as well as improvements to the zirconium redistribution model through implementation of a fully consistent phase diagram.

Although the currently implemented swelling model has already been identified as a weak spot in metallic fuels modeling in BISON, the difficulty in developing a predictive model, a problem shared across all non-empirical fission gas models, has
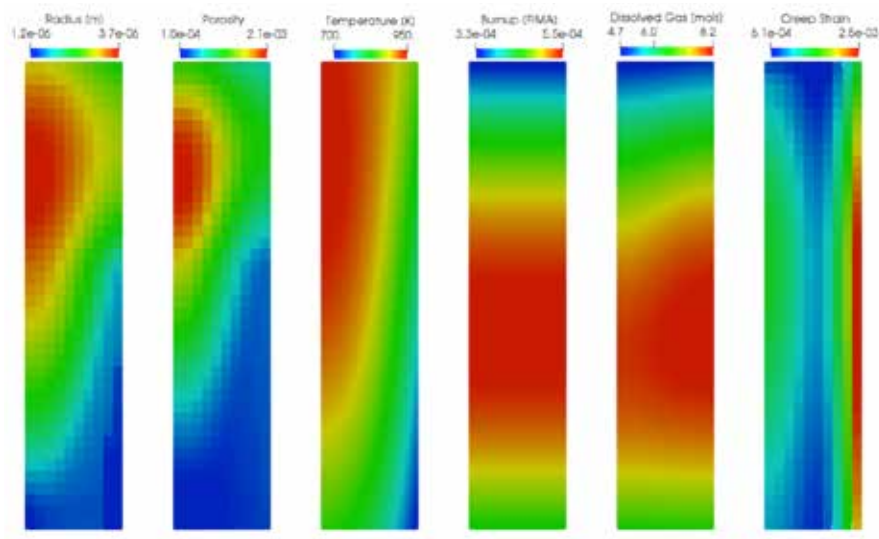

Figure 1. Example results from the new fission gas swelling and release model.

hindered past development. In FY17, a new model based on bubble surface force equilibrium was proposed, and implemented without great success due to limitations imposed by the underlying MOOSE mechanical solvers as well as incomplete formulations of the swelling model itself. These hurdles were overcome through a complete fleshing out of the swelling model, including approximations where necessary (Matthews, Los Alamos National Laboratory (LANL). In addition, the fully consistent tangent operator for creep, swelling, and creep with swelling was derived to aid in convergence (Prakash, LANL). The creep formulation was implemented into MOOSE for the benefit of all users (Matthews, LANL), and was shown to significantly improve convergence and reduce run times by an order of magnitude in problems with large creep strain (e.g. metallic fuel, cladding). Full implementation of the swelling model is still underway as the last numerical problems are being overcome, although initial results show favorable comparisons with fission gas release data.

Zirconium redistribution in metallic fuels has historically relied on a simplified pseudo-binary phase diagram to calculate the individual phase contribution to the total diffusivity. Approximations such as reduced number of phases and horizontal temperature transitions bake in potentially unphysical behaviors into the species redistribution solver. This year, we implemented an extensive set of phase information lookup tables formulated directly via the free energy formulation of U-Zr and U-Pu-Zr fuel (Hirschhorn, LANL/University of Florida (UF). This enables direct use of physically representative phase fractions and compositions, removing an unnecessary layer of simplification via the simplified phase diagram. Using a newly derived zirconium dependent fission rate formulation that includes burnup dependent parameters (Wilkerson, LANL), a new calibration is currently underway to try and bridge the gap between differing data sets (Stevens, LANL).
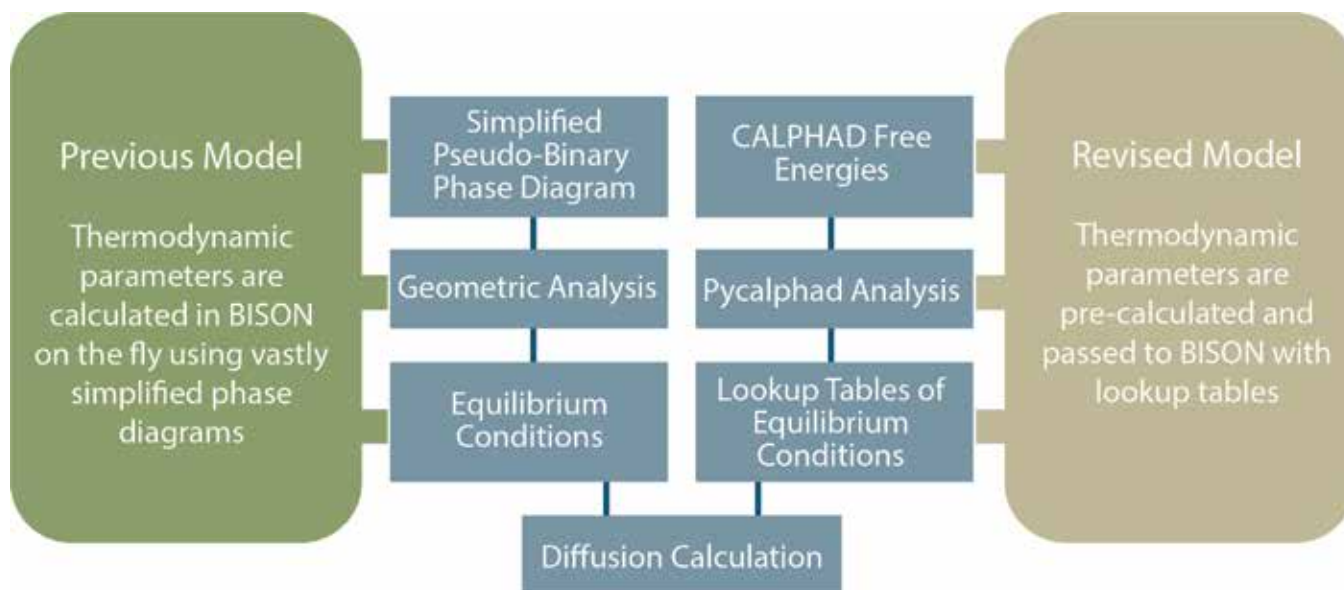
Figure 2. Comparison between the previous and updated model for implementing phase information into the BISON zirconium redistribution model.
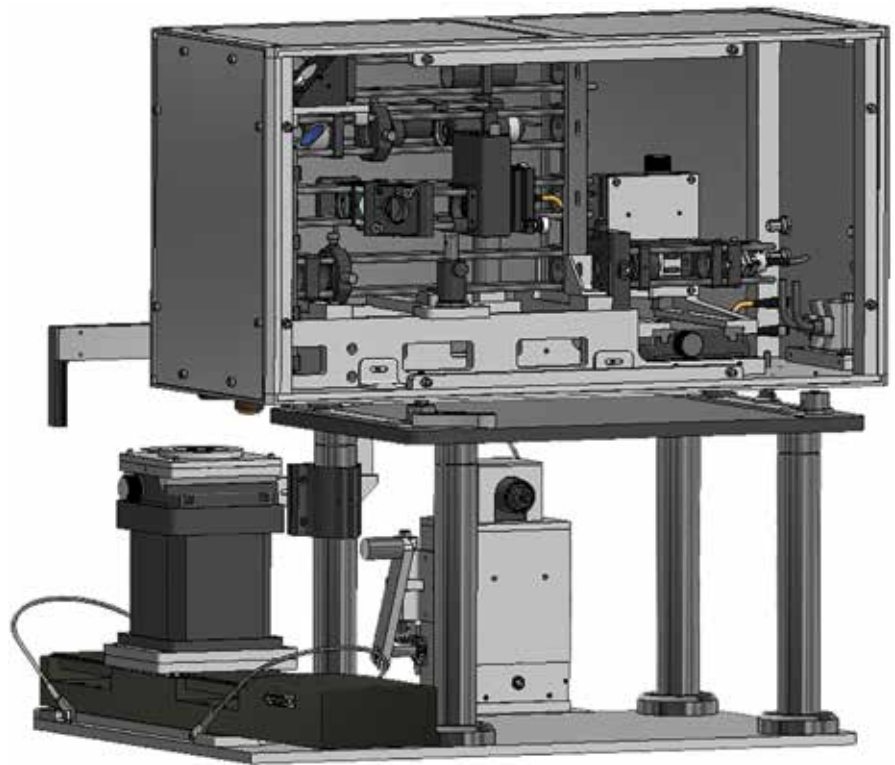
3.5 CAPABILITY DEVELOPMENT

Advanced Instrument Development: Thermal Conductivity Microscope

Principal Investigator: David Hurley
Collaborators: Robert Schley

Figure 1. 3D model of Thermal Conductivity Microscope in cell system

The Thermal Conductivity Microscope (TCM) is an instrument designed to measure thermal property of irradiated samples. The TCM simultaneously measures the thermal diffusivity and conductivity using modulated thermoreflectance. This measurement approach involves measuring the temperature field spatial profile of samples excited by an amplitude modulated continuous wave laser beam. A thin gold film is applied to the samples to ensure strong optical absorption and to establish a second boundary condition that introduces an expression containing the substrate thermal conductivity. The diffusivity and conductivity are obtained by comparing the measured phase profile of the temperature field to a continuum based model. The TCM has been designed to operate in a radiation hot cell environment. It can be controlled remotely via the software interface and sample loading is compatible with hot cell manipulators.

Project Description

The TCM will provide a new capability for the measurement of thermal properties of irradiated samples. Thermal conductivity and thermal diffusivity are among the most important properties of a nuclear fuel, but they can also be one of the most difficult to measure. The TCM provides micron-level thermal property information that is commensurate with microstructure heterogeneity. The development of the TCM connects closely with INL's larger PIE effort to provide new validation metrics for fundamental computational material science models.

A prototype TCM instrument was designed and tested during FY16. Work for FY 17 centered on developing ancillary equipment to support standalone remote operation in preparation for installation in the Irradiated Materials Characterization Laboratory (IMCL) and initiation of equipment qualification. The primary tasks included, 1- development of cell feedthroughs for both optical and electrical signals, 2 – design and construction of a film thickness monitor, 3 – development of an equipment qualification plan, 4 – develop an integrated equipment rack to house lasers and instrumentation, and 5 – develop coding instrumentation software for controlling the TCM.

Accomplishments

Several accomplishments were realized in the development of the TCM during FY17. Using the standard glove box panel, a layout for the electrical and optical feedthroughs was developed. Feedthroughs were incorporated into standard vacuum fittings to meet leak rate requirements and for ease of installation and replacement if necessary. Vendors for the feedthroughs were identified and the fittings were procured and tested.

The ability of the TCM to directly measure thermal conductivity requires a thin gold film to be applied to the sample. The film thickness is an important parameter that must be measured. A unique film thickness monitor was developed which allows film thickness to be determined based on light transmission. This device was designed and fabricated for use in a hot cell using remote manipulators for sample placement.

The TCM provides micron-level thermal property information that is commensurate with microstructure heterogeneity. The TCM will allow for a direct comparison between microstructure and material properties without the need to upscale modeling results to length scales associated with current experimental capability. The development of the TCM connects closely with INL's larger PIE effort to provide new validation metrics for fundamental computational material science models.

Figure 2. Thermal Conductivity Microscope testing at the Materials and Fuels Complex.

qualification at the MFC Mockup wall will be performed to verify the equipment can be operated in a hot cell environment with manipulators.

In addition to the thickness monitor and EQP, an equipment rack and software were also developed. The equipment rack incorporates all of the instrumentation and controls for remotely operating the TCM. An interface box in conjunction with a newly developed power box allow automation of signal switching and powering of the detectors, sensors, cameras and LEDs necessary for TCM operation. A method for sample height determination which will prevent sample to objective interference was also developed and incorporated. A graphical user interface (GUI) was developed using MATLAB to provide point-and-click control of the TCM and thickness monitor, as well as data analysis.

The TCM and associated equipment were entered into stage one mockup at MFC on 9/6/2017. This included the TCM, the equipment rack and all ancillary equipment as well as the film thickness monitor. Mockup testing is underway and includes manipulator compatibility and performance verification test using thermal conductivity standards.

An equipment qualification plan (EQP) for stage I/II mockup was also completed in FY17. This document contains basic instructions for the proper assembly of the TCM in the Mockup Shop located at the Materials and Fuels Complex (MFC) at Idaho National Laboratory (INL). Phase I: The TCM will be setup on the base plate, the electro-mechanical components positioned, and all electrical wiring will be connected. Phase II: After the initial system operation is verified, remote

APPENDIX

4.1 PUBLICATIONS

Author (s)	Title	Publication
J.L. Arndt, E.J. Lahoda and R.L. Oelrich	Westinghouse Accident Tolerant Fuel and Its Benefits for CANDU Reactors	Proceedings of the 38th Annual Conference of the Canadian Nuclear Society, Saskatoon, SK, Canada, Jun 3 - 6, 2018
E. Aydogan, O. El-Atwani, S. Takajo, S. C. Vogel and S. A. Maloy	High Temperature Microstructural Stability and Recrystallization Mechanisms in 14YWT Alloys	Acta Materialia 148 (2018), 467-481
M.T. Benson, J.A. King, R.D. Mariani, and M.C. Marshall	SEM Characterization of Two Advanced Fuel Alloys: U-10Zr-4.3Sn and U-10Zr-4.3Sn-4.7Ln	Journal of Nuclear Materials 494 (2017), 334-341
M.T. Benson, J.A. King, and R.D. Mariani	Investigation of Tin as a Fuel Additive to Control FCCI	Proceedings of the Minerals, Metals & Materials Society (TMS), F12 (2018), 695-702
M.T. Benson, L. He, J.A. King, and R.D. Mariani	Microstructural Characterization of Annealed U-12Zr-4Pd and U-12Zr-4Pd-5Ln: Investigating Pd as a Metallic Fuel Additive	Journal of Nuclear Materials 502 (2018), 106-112
M.T. Benson, L. He, J.A. King, R.D. Mariani, A. Winston, and J. Madden	Microstructural Characterization of As-cast U-20Pu-10Zr-3.86Pd and U-20Pu-10Zr-3.86Pd-4.3Ln	Journal of Nuclear Materials 508 (2018), 310-318
M.T. Benson, Y. Xie, J.A. King, K.R. Tolman, R.D. Mariani, I. Charit, J. Zhang, M.P. Short, S. Choudhury, R. Khanal, and N. Jerred	Characterization of U-10Zr-2Sn-2Sb and U-10Zr-2Sn-2Sb-4Ln to Assess Sn+Sb as a Mixed Additive System to Bind Lanthanides	Journal of Nuclear Materials 510 (2018), 210-218
J. Bischoff, C.Delafoy, C. Vauglin, P. Barberis, C. Roubeyrie, D. Perche, and K. Nimishakavi	AREVA NP's Enhanced Accident-tolerant Fuel Developments: Focus on Cr-coated M5 Cladding	Nuclear Engineering and Technology 50(2), (2018), 223-228
N. Capps, A. Mai, M. Kennard and W. Liu	PCI Analysis of Zircaloy-coated Clad Under LWR Steady State and Reactor Startup Operations Using BISON Fuel Performance Code	Nuclear Engineering and Design, 332 (2018), 383-391
U. Carvajal-Nunez et al.	Determination of Elastic Properties of Polycrystalline U_3Si_2 Using Resonant Ultrasound Spectroscopy	Journal of Nuclear Materials, 498 (2018), 438-444

| Author (s)	Title	Publication
U. Carvajal-Nunez et al.	Mechanical Properties of Uranium Silicides by Nanoindentation and Finite Elements Modeling	The Journal of The Minerals, Metals & Materials Society 70 (2018), 203-208
Y. Che, G. Pastore, J. Hales and K. Shirvan	Modeling of Cr_2O_3-doped UO_2 as a Near-term Accident Tolerant Fuel for LWRs using the BISON Code	Nuclear Engineering and Design, 337 (2018), 271-278
R. Chipaux, G. Cecilia, M. Beauvy, and R. Troc	Capacite thermique a haute temperature de UBe_{13}, $ThBe_{13}$ et UB_4*	Journal Less-Common Met., 121 (1986), 347–351
M.N. Cinbiz, N. R. Brown, R. R. Lowden, M. Gussev, K. Linton, and K. A. Terrani	Report on Design and Failure Limits of SiC/SiC and FeCrAl ATF Cladding Concepts under RIA	ORNL/LTR-2018/521 NT-M3FT-2018-204032, Oak Ridge, 2018
A. Csontos, J.Whitt, J. Carmack, M. Gavrilas, J. Williams, and E. Lahoda	Panel Discussion On ATF Implementation In The Nuclear Industry	Proceedings of the American Nuclear Society (ANS) Meeting, Philadelphia, PA, Jun 18-21, 2018.
P. Dahl, I. Kaus, Z. Zhao, M. Johnsson, and M. Nygren	Densification and properties of zirconia prepared by three different sintering techniques	vol. 33, pp. 1603–1610, 2007
J. P. Dancausse et al	Compression study of uranium borides UB_2 , UB_4 and UB_{12} by synchrotron X-ray diffraction	Journal of Alloys Compd., 189 (1992), 205–208
C.P. Deck, H.E. Khalifa, G. M. Jacobsen, J. Shedder, J. Zhang, C. Bacalski, J.G. Stone, Chunghao Shih, Gokul Vasudevamurthy, Edward Lahoda, and Christina A. Back	Development of Engineered SiC-SiC Accident Tolerant Fuel Cladding	Proceedings of the 42nd International Conference and Expo on Advanced Ceramics and Composites, Daytona Beach, Florida, Jan 23, 2018
C.P. Deck, H.E. Khalifa, G. Jacobsen, J. Sheeder, K. Shapovalov, S. Gonderman, E. Song, J. Gazza, C.A. Back, P. Xu, F. Boylan, and R. Jacko	Demonstration Of Engineered Multi-Layered Sic-Sic Cladding With Enhanced Accident Tolerance	Proceedings of the TopFuel 2018 Conference, Prague, Czech Republic, Sep 30 – Oct 4, 2018
M. Demuynck, J. Erauw, O. Van Der Biest, F. Delannay, and F. Cambier	Densification of alumina by SPS and HP: A comparative study	Journal of Eur. Ceram. Soc., 32 (2012), 1957–1964
Y. Deng, K. Shirvan, Y. Wu and G.Su	Probabilistic View of SiC/SiC Composite Cladding Failure Based on Full Core Thermo-mechanical Response	Journal of Nuclear Materials 507 (2018), 24-37

Author (s)	Title	Publication
S. Dryepondt, K.A. Unocic, D.T. Hoelzer, C.P. Massey and B.A. Pint	Development of Low-Cr ODS FeCrAl Alloys for Accident-tolerant Fuel Cladding	Journal of Nuclear Materials 501 (2018), 59-71
S. Dryepondt, C.P. Massey, M.N. Gussev, K.D. Linton and K.A. Terrani	Tensile Strength and Steam Oxidation Resistance of ODS FeCrAl Sheet and Tubes	ORNL Report TM-2018/870 (2018)
J. Fink	Thermophysical properties of uranium dioxide	Journal of Nuclear Materials 279 (1) 1-18 (200)
F. Franceschini, J. King, E. Lahoda, R. Oelrich, and S. Ray	Implementation of Westinghouse ATF to Extend Cycle Length of Current PWRs to 24-month Cycles	Proceedings of the PHYSOR 2018 Conference, Cancun, Mexico, Apr 22-26, 2018
D. R. Fredrickson, M. G. Barner, R.D., Chasanov, R. L. Nuttall, R. Kleb, and W. N. Hubbard	The Enthalpy of Uranium Diboride from 600 to 1500° K by Drop Calorimetry	High Temp. Sci. 1, vol. 80, pp. 373–380, 1969.
D. R. Fredrickson, R. D. Barnes, and M. G. Chasanov, "	Enthalpy of uranium diboride from 1300 to 2300˚K by drop calorimetry	High Temp. Sci., vol. 2, pp. 299–301, 1970.
K. Gofryk	Effect of Off-stoichiometry and Grain Size on Thermal Transport in Uranium Dioxide	Talk given at the Nuclear Materials conference NuMat 2018, Seattle, WA, 2018
A.Gurgen and K. Shirvan	Estimation of Coping Time in Pressurized Water Reactors for Near Term Accident Tolerant Fuel Claddings	Nuclear Engineering and Design, 337 (2018), 38-50
E. Jossou, L. Malakkal, B. Szpunar, D. Oladimeji, and J. A. Szpunar	A First Principles Study of the Electronic Structure, Elastic and Thermal Properties of UB_2	Journal of Nuclear Matter 490 (2017), 41-48
Z. Karoutas, W. Luangdilok, M. Shockling, R. Schneider, E. Lahoda and P. Xu	Update On Westinghouse Benefits of Encore® Fuel	Proceedings of the TopFuel 2018 Conference, Prague, Czech Republic, Sep 30 - Oct 4, 2018
T. Koyanagi, Y. Katoh, G. Singh, C. Petrie, C. Deck, K. Terrani1; T. Koyanagi, Y. Katoh, G. Singh, C. Petrie, C. Deck and K. Terrani	Post-irradiation Examination of SiC Tubes Neutron Irradiated Under a Radial High Heat Flux	Proceedings of the 42nd International Conference and Expo on Advanced Ceramics and Composites, Daytona Beach, Florida, January 23rd, 2018

Author (s)	Title	Publication
E. Lahoda	Approaches for Accelerating Licensing of ATF Products	American Nuclear Society presentation, Washington, D.C., November 1, 2017
E. Lahoda	Westinghouse Accident Tolerant Fuel Materials	Materials Science and Technology Meeting, Pittsburgh, PA., October 10, 2017
Y.-P. Lin, R. M. Fawcett, S. Desilva, D. R. Luz, M. O. Yilmaz, P. Davis, R. Rand, P. E. Cantonwine, R. B. Rebak, R. Dunavant and N. Satterlee	Path Towards Industrialization of Enhanced Accident Tolerant Fuel	Proceedings of the TopFuel 2018 Conference, Prague, Czech Republic, Sep 30-Oct 4, 2018
Y. Long*, P.J. Kersting, O. Linsuain, T.M. Crede and R. L. Oelrich	Fuel Performance Analysis of EnCore® Fuel	Proceedings of the TopFuel 2018, Conference, Prague, Czech Republic, Sep 30 – Oct 4, 2018
B.R. Maier, H. Yeom, G.O. Johnson, T. Dabney, J. Walters, J. Romero, H. Shah, P. Xu, and K. Sridharan	Development of Cold Spray Coatings for Accident-Tolerant Fuel Cladding in Light Water Reactors	Journal of Minerals, Metals, and Materials Society (JOM), 70(2) (2018), 198
C.P. Massey, K.A. Terrani, S. Zinkle, P Edmondson and S. Dryepondt	Influence of Mechanical Alloying and Extrusion Conditions on the Microstructure and Tensile Properties of Low-Cr ODS FeCrAl Alloys	Journal of Nuclear Materials, accepted for publication (2018)
C. Matthews, G. Stevens, and C. Unal	Calibration of Zr Redistribution Models for Metallic Fuel in BISON	Transactions of the American Nuclear Society Annual Meeting, Philadelphia, PA, Jun 17-21, 2018
J. W. McMurray, J. O. Kiggans, G. W. Helmreich and K. A. Terrani	Production of Near Full Density Uranium Nitride Microspheres with a Hot Isostatic Press	Journal of American Ceramic Society 101(10) (2018) 4492-4497
J. W. McMurray and T. M. Besmann	Thermodynamic Modeling of Nuclear Fuel Material	Handbook of Materials Modeling 2nd Edition, editors Wanda Andreoni and Sidney Yip (in press)
H. Muta, K. Kurosaki, M. Uno, and S. Yamanaka	Thermal and mechanical properties of uranium nitride prepared by SPS technique	Journal of Material Science 43(19) (2008), 6429–6434
R. Oelrich, P. Xu, E. Lahoda and C. Deck	Update on Westinghouse EnCore Accident Tolerant Fuel Program	Proceedings of the American Nuclear Society (ANS) Meeting, Philadelphia, PA, Jun 18-21, 2018

Author (s)	Title	Publication
R. Oelrich, S. Ray, Z. Karoutas, P. Xu, J. Romero, H.Shah, E. Lahoda and F. Boylan	Overview of Westinghouse Lead Accident Tolerant Fuel Program	Proceedings of the TopFuel 2018 Conference, Prague, Czech Republic, Sep 30 - Oct 4, 2018
Pal, S., M. E. Alam, S. A. Maloy, D. T. Hoelzer and G. R. Odette	Texture Evolution and Microcracking Mechanisms in As-extruded and Cross-rolled conditions of a 14YWT Nanostructured Ferritic Alloy	Acta Materialia 152 (2018) 338-357
P. Xu, E. Lahoda, F. Boylan and R.L. Oelrich	Status Update on Westinghouse EnCoreTM SiC/SiC Composite Cladding Development	Proceedings of the 42nd International Conference and Expo on Advanced Ceramics and Composites, Daytona Beach, FL, Jan 21-26, 2018
C.M. Petrie, J.R. Burns, R.N. Morris, K.R. Smith, A.G. Le Coq, and K.A. Terrani	Irradiation of Miniature Fuel Specimens in the High Flux Isotope Reactor	ORNL/SR-2018/844, Oak Ridge National Laboratory, Oak Ridge, TN (2018)
C.M. Petrie, J.R. Burns, R.N. Morris, and K.A. Terrani	Accelerated Irradiation Testing of Miniature Fuel Specimens	Transactions of the American Nuclear Society118 (2018), 1476-1479
C.M. Petrie, T. Koyanagi, R.H. Howard, K.G. Field, J.R. Burns, and K.A. Terrani	Accelerated Irradiation Testing of Miniature Nuclear Fuel and Cladding Specimens	Proceedings of the TopFuel 2018 Conference, Prague, Czech Republic, Sep 30 - Oct 4, 2018
A.M. Raftery, R.N. Morris, K.R. Smith, G.W. Helmreich, C.M. Petrie, K.A. Terrani and A.T. Nelson	Development of a Characterization Methodology for Post-irradiation Examination of Miniature Fuel Specimens	Oak Ridge National Laboratory, August 2018
R. B. Rebak	Versatile Oxide Films Protect FeCrAl Alloys Under Normal Operation and Accident Conditions in Light Water Power Reactors	JOM January 2018, doi/10.1007/s11837-017-2705-z.
R. B. Rebak, V. K. Gupta and M. Larsen	Oxidation Characteristics of Two FeCrAl Alloys in Air and Steam from 800°C to 1300°C	JOM August 2018, doi/10.1007/s11837-018-2979-9

Author (s)	Title	Publication
S. M. Scott, T. Yao, F. Lu, G. Xin, W. Zhu, and J. Lian	Fabrication of lanthanum-doped thorium dioxide by high-energy ball milling and spark plasma sintering	Journal of Nuclear Materials 485 (2017), 207–215
A. Seshadri, B. Philips and K. Shirvan	Towards Understanding the Effects of Irradiation on Quenching Heat Transfer	Journal of Heat and Mass Transfer 127 (2018) 1087-1095
A. Seshadri and K. Shirvan	Quenching Heat Transfer Analysis of Accident Tolerant Coated Fuel Cladding	Nuclear Engineering and Design 338 (2018), 5-15
M. Sevecek, A. Gurgen, A. Seshadri, Y. Che, M. Wagih, B. Phillips, V. Champagne and K. Shirvan	Development of Cr Cold-Sprayed Fuel Cladding with Enhanced Accident Tolerance	Nuclear Engineering and Technology Journal 50 (2018), 229-236
J. Sheeder, S. Gonderman, G. Jacobsen, H.E. Khalifa, C. Shih, E. Song, K. Shapovalov and C.P. Deck	Non-Destructive Evaluation of Sealed SiC-SiC Composite Cladding Structures Using X-Ray Computed Tomography, Pycnometry and Helium Leak Testing	Proceedings of the 42nd International Conference and Expo on Advanced Ceramics and Composites, Daytona Beach, FL, Jan 21-26, 2018
K. Shrestha, T. Yao, J. Lian, D. Antonio, M. Sessim, M. Tonks, and K. Gofryk	The Grain-size Effect on Thermal Conductivity of Uranium Dioxide	In preparation
B. Spencer et al.	Bison Improvements for Robustness and Speed	L2:FMC.P16.01, pp. 1–19,(2018)
L.N. Squires, J.A. King, R.S. Fielding and P. Lessing	Isolation of High Purity Americium Metal Via Distillation	Journal of Nuclear Materials 500 (2018), 26-32
Dr. K. Sridharan		Invited talk given by UW at the Metallurgical Society (TMS) annual meeting in March 2018.
S. Ray	The Need for Hot Cells for Nuclear R&D - The Role of Hot Cells in New Fuel Development	American Nuclear Society presentation, Washington, D.C., October 31, 2017
C. Unal, G.N. Stevens, and C. Matthews	Progressive Bayesian Calibration of the BISON Fuel Performance Capability	Proceedings of the TopFuel 2018 Conference, Prague, Czech Republic, Sep 30 – Oct 4, 2018
M. Wagih, B. Spencer, J. Hales and K. Shirvan	Fuel Performance of Chromium-coated Zirconium Alloy and Silicon Carbide Accident Tolerant Fuel Claddings	Annals of Nuclear Energy, 120 (2018), 304-318

Author (s)	Title	Publication
J. Wang, H.J. Jo and M.L. Corradini	Potential Recovery Actions from a Severe Accident in a PWR: MELCOR Analysis of a Station Blackout Scenario	Nuclear Technology, Online, 2018. DOI: 10.1080/00295450.2018.1464838
J. T. White	Report on the basic chemistry , microstructure and physical properties of high uranium density boride compounds	2015
P. Xu, E.J. Lahoda, J. Lyons, C.P. Deck and G.E. Kohse	Status Update on Westinghouse Sic Composite Cladding Fuel Development	Proceedings of the TopFuel 2018 Conference, Prague, Czech Republic, Sep 30 – Oct 4, 2018
T. Yao, S. M. Scott, G. Xin, and J. Lian	TiO_2 doped UO_2 fuels sintered by spark plasma sintering	Journal of Nuclear Materials 469 (2016), 251–261
S. Yeo, E. McKenna, R. Baney, G. Subhash, and J. Tulenko	Enhanced thermal conductivity of uranium dioxide-silicon carbide composite fuel pellets prepared by Spark Plasma Sintering (SPS)	Journal of Nuclear Materials. 433 (1-3) (2013), 66–73
H. Yeom, T. Dabney, G. Johnson, B. Maier, M. Lenling and K. Sridharan	Improving Deposition Efficiency in Cold Spraying Chromium Coatings by Powder Annealing	The International Journal of Advanced Manufacturing Technology, submitted, 2018.
H. Yeom, B. Maier, G. Johnson, T. Dabney, J. Walters and K. Sridharan	Development of Cold Spray Process for Oxidation-Resistant FeCrAl and Mo Diffusion Barrier Coatings on Optimized ZIRLO®*	Journal of Nuclear Materials. 507 (2018) 306–315
A. Zalkin and D. H. Templeton	The crystal structures of CeB4, ThB4 and UB4	Acta Crystallogr 6 (3) (1953), 269–272
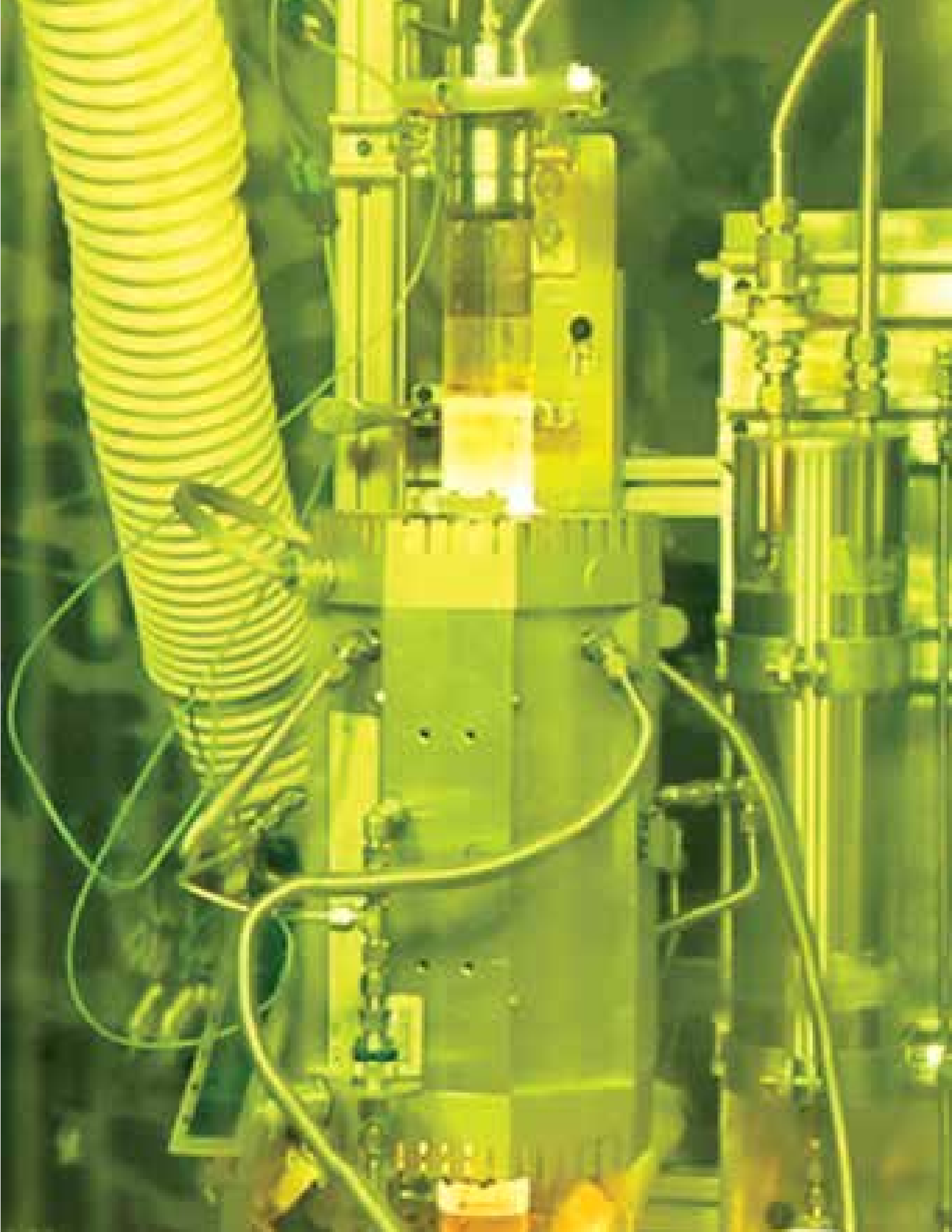
4.2 FY-17 LEVEL 2 MILESTONES

Work Package Title	Site	Work Package Manager	Level 2 Milestone
AFC Campaign Management - INL	INL	Beverly, Ed	Provide Draft Update of the ATF Roadmap to Congress including input from ATF Industry Teams
ATF-1 Irradiation Testing in ATR - INL	INL	Core, Greg	Issue a report documenting the development of the NDMAS ATF-2 Database and verification of data quality
ATF-2 ATR Loop Irradiation - INL	INL	Hoggard, Gary	Assemble ATF-2 and ensure quality inspected test train is ready for transfer to ATR for insertion
ATF Transient Irradiation Testing - INL		Dempsey, Doug	Report on results of the initial TREAT Transient Prescription Testing and Analysis
Static Capsule Irradiation Device Prototype - INL	INL	Pavey, Todd	Complete design and build experiment support equipment to support instrumentation and controls for each experiment
Qualify Baseline TREAT Experiment Instrumentation - INL	INL	Pavey, Todd	Issue a summary technical report for instrumentation research, development, and demonstration for the transient testing
ATF PIE - INL	INL	Harp, Jason	Issue a report documenting ATF PIE results
Static Capsule Irradiation Device Prototype - INL	INL	Pavey, Todd	Issue a summary technical report for conceptual design of MARCH-SERTTA
ATF Transient Irradiation Testing - INL	INL	Dempsey, Doug	Complete ATF-SETH testing in TREAT
Fuel Characterizations - INL	INL	Papesch, Cynthia	Issue update of Metallic Fuels Handbook
Advanced Fabrication Development - INL	INL	Fielding, Randy	Demonstrate fabrication of annular U-Zr fuel by extrusion
Feedstock Preparation/Purification - INL	INL	Squires, Leah	Issue report on status of Am and Np purification processes and future directions
PIE and Analyses - INL	INL	Harp, Jason	Issue PIE Report on AFC-3C, AFC-3D, and AFC-4A
Fuel Modeling Support - INL	INL	Medvedev, Pavel	Issue status report on BISON development and applications for transmutation fuels
Advanced Instrument Development and BRR - INL	INL	Hurley, Dave	Closeout Phase II Qualification of the Thermal Conductivity Microscope for IMCL
Advanced Ceramic Fuel Performance and Qualification -LANL	LANL	McClellan, Ken	Fabricate ceramic minifuel samples and ensure they are ready to ship from LANL to HFIR for irradiation

Work Package Title	Site	Work Package Manager	Level 2 Milestone
Advanced Ceramic Fuel Development - LANL	LANL	McClellan, Ken	Issue a report on the determination of the feasibility to 'waterproof' UN for LWR use
High Dose Materials Testing for FR -LANL	LANL	Saleh, Tarik	Perform Testing and Issue report on tensile properties of high dose irradiated AR cladding materials
Irradiation Testing of Advanced Fuels and Cladding Materials - ORNL	ORNL	Terrani, Kurt	Initiate miniature fuel irradiation in support of separate effects testing
ATF cladding tests under Design Basis Accidents - ORNL	ORNL	Linton, Kory	Issue report on In-Cell Re-fabrication of Irradiated rods and LOCA Testing
ATF SiC Cladding and Core Component Development	ORNL	Katoh, Yutai	Issue Update to LWR SiC/SiC Cladding Handbook of Properties
ATF FeCrAl Cladding Development and Testing	ORNL	Yamamoto, Yuki	Issue Update FeCrAl cladding handbook of properties
Microstructure Analysis and Evolution in Ceramic Fuels (ORNL)	ORNL	Nelson, Andy	Issue report on the development of a characterization methodology for separate effects testing of irradiated ceramic fuels

4.3 AFC NEUP GRANTS

Active Projects Awarded in 2014

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
University of Notre Dame	Assessment of Corrosion Resistance of Promising Accident Tolerant Fuel Cladding Under Reactor Conditions	David Bartels
Virginia Polytechnic Institute and State University	Thermal Conductivity in Metallic Fuels	Celine Hin
Virginia Polytechnic Institute and State University	SiC-ODS Alloy Gradient Nanocomposites as Novel Cladding Materials	Kathy Lu

Active Projects Awarded in 2015

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
Northwestern University	Electrically-Assisted Tubing Processes for Enhancing Manufacturability of Oxide Dispersion Strengthened Structural Materials for Nuclear Reactor Applications	Jian Cao
Massachusetts Institute of Technology	Multilayer Composite Fuel Cladding for LWR Performance Enhancement and Severe Accident Tolerance	Michael Short
University of Wisconsin, Madison	Radiation-Induced Swelling and Micro-Cracking in SiC Cladding for LWRs	Izabela Szlufarska
University of California, Berkeley	Developing a Macro-Scale SiC-cladding Behavior Model Based on Localized Mechanical and Thermal Property Evaluation on Pre- and Post-irradiation SiC-SiC Composites.	Peter Hosemann
Massachusetts Institute of Technology - IRP	Development of Accident Tolerant Fuel Options for Near Term Applications	Jacopo Buongiorno
University of Tennessee at Knoxville	Radiation Effects on High Thermal Conductivity Fuels	Steven Zinkle

Active Projects Awarded in 2016

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
University South Carolina	Phase Equilibria and Thermochemistry of Advanced Fuels: Modeling Burnup Behavior	Theodore Besmann
North Carolina State University	Microstructure Experiments-Enabled MARMOT Simulations of SiC/SiC-based Accident Tolerant Nuclear Fuel System	Jacob Eapen
Purdue University	Microstructure, Thermal, and Mechanical Properties Relationships in U and UZr Alloys	Maria Okuniewski
Pennsylvania State University	A Coupled Experimental and Simulation Approach to Investigate the Impact of Grain Growth, Amorphization, and Grain Subdivision in Accident Tolerant U_3Si_2 Light Water Reactor Fuel	Michael Tonks
University of Idaho - IRP	A Science Based Approach for Selecting Dopants in FCCI-Resistant Metallic Fuel Systems	Indrajit Charit
The Ohio State University	Alloying Agents to Stabilize Lanthanides Against Fuel Cladding Chemical Interaction: Tellurium and Antimony Studies	Jinsuo Zhang

Active Projects Awarded in 2017

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
University of Wisconsin-Madison	Extreme Performance High Entropy Alloys (HEAs) Cladding for Fast Reactor Applications	Adrien Couet
University of Wisconsin-Madison	Critical Heat Flux Studies for Innovative Accident Tolerant Fuel Cladding Surfaces	Michael Corradini
Colorado School of Mines	Development of Advanced High-Cr Ferritic/ Martensitic Steels	Kester Clarke
Massachusetts Institute of Technology	Determination of Critical Heat Flux and Leidenfrost Temperature on Candidate Accident Tolerant Fuel Materials	Matteo Bucci
University of New Mexico	An Experimental and Analytical Investigation into Critical Heat Flux (CHF) Implications for Accident Tolerant Fuel (ATF) Concepts	Youho Lee
Missouri University of Science and Technology	Gamma-ray Computed and Emission Tomography for Pool-Side Fuel Characterization	Joseph Graham
Virginia Commonwealth University	Evaluation of Accident Tolerant Fuels Surface Characteristics in Critical Heat Flux Performance	Sama Bilbao y Leon
University of New Mexico	Nanostructured Composite Alloys for Extreme Environments	Osman Anderoglu

Active Projects Awarded in 2018

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
University of California, Berkeley	Understanding of degradation of SiC/SiC materials in nuclear systems and development of mitigation strategies	Peter Hosemann
University of Minnesota, Twin Cities	Probabilistic Failure Criterion of SiC/SiC Composites Under Multi-Axial Loading	Jialiang Le
University of Wisconsin-Madison	Advanced Coating and Surface Modification Technologies for SiC-SiC Composite for Hydrothermal Corrosion Protection in LWR	Kumar Sridharan
University of Michigan	Mechanistic Understanding of Radiolytically Assisted Hydrothermal Corrosion of SiC in LWR Coolant Environments	Gary Was
University of Florida	Multiaxial Failure Envelopes and Uncertainty Quantification of Nuclear-Grade SiCf/SiC Woven Ceramic Matrix Tubular Composites	Ghatu Subhash
University of Notre Dame	Radiolytic Dissolution Rate of Silicon Carbide	David Bartels
University of South Carolina	Development of Multi-Axial Failure Criteria for Nuclear Grade SiCf-SiCm Composites	Xinyu Huang

Lead University	Title	Principle Investigator
University of California, Berkeley	Bridging the length scales on mechanical property evaluation	Peter Hosemann
Purdue University	Microstructure-Based Benchmarking for Nano/Microscale Tension and Ductility Testing of Irradiated Steels	Janelle Wharry
University of Utah	Benchmarking Microscale Ductility Measurements	Owen Kingstedt
University of Nebraska, Lincoln	Bridging microscale to macroscale mechanical property measurements and predication of performance limitation for FeCrAl alloys under extreme reactor applications	Jian Wang
Virginia Polytechnic Institute and State University	C-SiOC-SiC Coated Particle Fuels for Advanced Nuclear Reactors	Kathy Lu
University of Tennessee at Knoxville	A novel and flexible approach for converting LWR UNF fuel into forms that can be used to fuel a variety of Gen-IV reactors	Craig Barnes
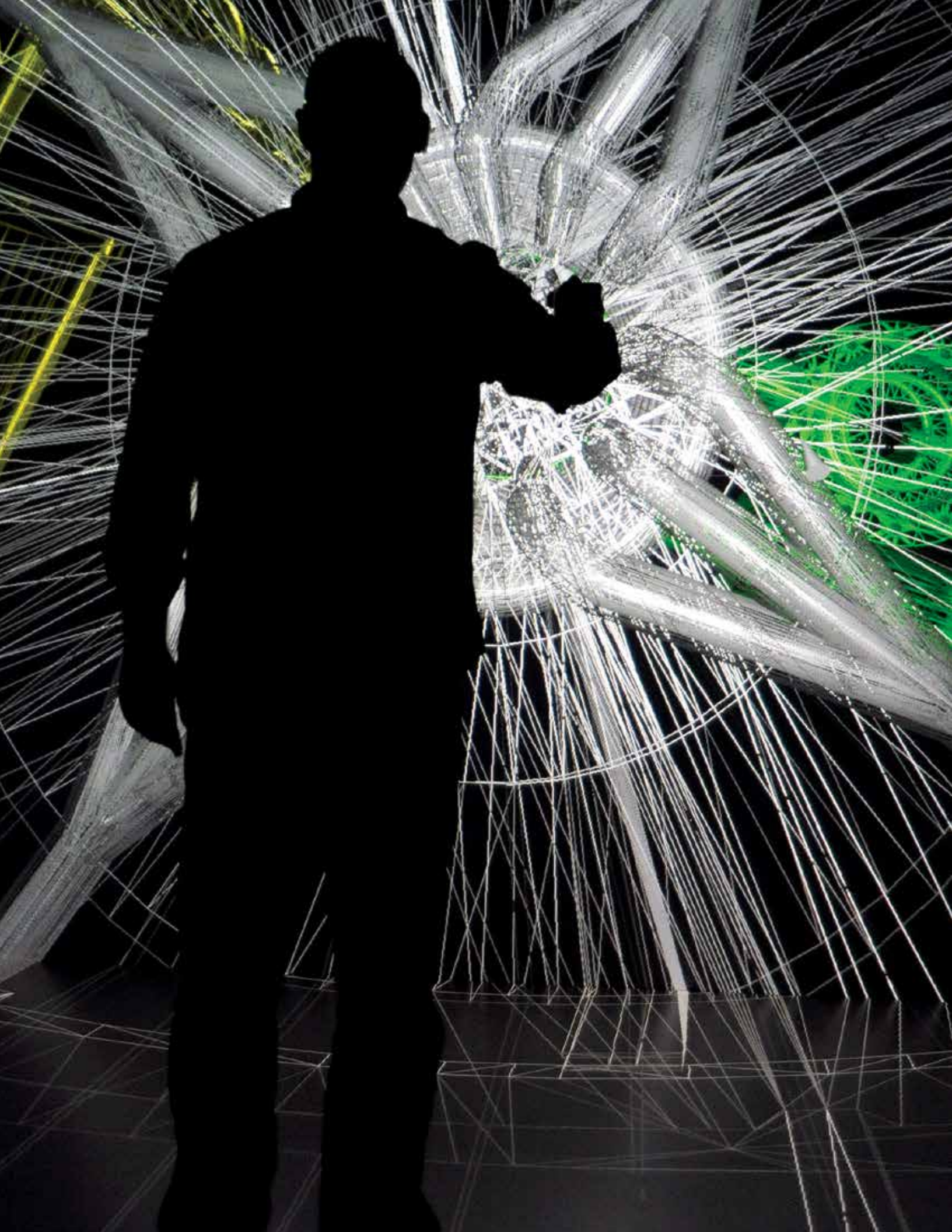
4.5 ACRONYMS

AFC..Advanced Fuels Campaign

AM..Additive Manufacturing

APT... Atom Probe Tomography

ANL ....Argonne National Laboratory

ANS .... American Nuclear Society

ARL.. Army Research Lab

ATF .... Accident Tolerant Fuel

ATR.. Advanced Test Reactor

BCC..Body Centered Cubic

BDBA...Beyond Design Basis Accident

BWR...Boiling Water Reactor

BNL .... Brookhaven National Laboratory

CEA .... Commissariat à l' Énergie Atomique

CTRN ....Carbothermic Reduction and Nitriding

CWRU .... Case Western Reserve University

DAS...Data Acquisition System

DBA .... Design Basis Accident

DED .... Directed Energy Deposition

DOE... Department of Energy

EATF ....Enhanced Accident Tolerant Fuel

EBR...Experimental Breeder Reactor

EBSD..Electron Backscatter Diffraction

ECCS .... Emergency Core Cooling System

ECR..Equivalent Clad Reacted

EDS ....Energy Dispersive X-Ray Spectroscopy

EFF..Experimental Fuel Facility

EGATFL...Expert Group on Accident Tolerant Fuel

AFC Advanced Fuels Campaign